**LICENSE RENEWAL INTERIM STAFF GUIDANCE**

**LR-ISG-2016-01**

**CHANGES TO AGING MANAGEMENT GUIDANCE FOR VARIOUS STEAM GENERATOR COMPONENTS**

## INTRODUCTION

This license renewal interim staff guidance (LR-ISG), LR-ISG-2016-01, "Changes to Aging Management Guidance for Various Steam Generator Components," describes changes to NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2 (December 2010), and NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), Revision 2 (December 2010).

This LR-ISG revises GALL Report aging management program (AMP) XI.M19, "Steam Generators," and SRP-LR Sections 3.1.2.2.11 and 3.1.3.2.11, "Cracking due to Primary Water Stress Corrosion Cracking." Specifically, this LR-ISG addresses the changes to aging management guidance on: (a) cracking due to primary water stress corrosion cracking (PWSCC) of divider plate assemblies and tube-to-tubesheet welds, and (b) loss of material due to boric acid corrosion of steam generator heads and tubesheets. In addition, changes are made to the associated aging management review (AMR) summary tables in the SRP-LR and the AMR tables in the GALL Report. This LR-ISG also revises the Final Safety Analysis Report (FSAR) supplement for the AMP that is documented in SRP-LR, Table 3.0-1, "FSAR Supplement for Aging Management of Applicable Systems."

These changes provide one acceptable approach for managing the associated aging effects for steam generator components within the scope of the License Renewal Rule (Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"). A licensee may cite LR-ISG-2016-01 in its license renewal application (LRA) until the guidance in this LR-ISG is incorporated into the license renewal guidance documents (i.e., GALL Report and SRP-LR). The guidance in this LR-ISG may be used as a basis for completion of license renewal commitments related to aging management for steam generator components.

The staff also plans to consider the information in this LR-ISG and make corresponding changes when finalizing the aging management guidance for the subsequent license renewal period (i.e., up to 80 years of operation), which is documented in draft NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal Report" (GALL-SLR), and draft NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal for Nuclear Power Plants" (SRP-SLR), if it is practicable to do so in terms of the development schedule for subsequent license renewal guidance.

**DISCUSSION**

Since the development of the NRC staff (hereafter staff) position on AMPs for divider plate assemblies and tube-to-tubesheet welds documented in Revision 2 of the GALL Report and SRP-LR, there has been additional operating experience, analyses, and testing related to these components (References 1-4). The additional analyses, performed by the industry, assessed the significance of potential cracks in the divider plate assemblies and their potential propagation to reactor coolant pressure boundary components such as tube-to-tubesheet welds and the channel head. The additional testing included studying the chemical composition of tube-to-tubesheet welds to assess their susceptibility to PWSCC.

In addition, degradation of the steam generator channel head and tubesheet has been observed at operating reactors. NRC Information Notice 2013-20, "Steam Generator Channel Head and Tubesheet Degradation" summarizes related operating experience (Reference 5).

Evaluation of PWSCC in Divider Plate Assemblies

In determining the degree of aging management needed to manage the effects of PWSCC for divider plate assemblies, the staff considered the susceptibility of divider plate materials to PWSCC and the safety significance of any PWSCC should it occur in the component.

PWSCC can initiate under specific conditions and chromium content in high-strength nickel alloys such as Alloy 600 and Alloy 690, and their associated weld materials. However, operating experience has shown that in the operating conditions (environment and stresses) of pressurized water reactors, Alloy 690 is highly resistant to PWSCC. This is due to the high chromium content of Alloy 690 in the range of approximately 27 to 31 percent. In comparison, Alloy 600 has a lower chromium content (approximately 14 to 17 percent) than Alloy 690 and is typically considered susceptible to PWSCC.

A few steam generators in foreign pressurized water reactors have exhibited PWSCC in divider plate assemblies. All but one of these instances of PWSCC have been observed in the divider plate assemblies that are approximately 1.3 inches thick (Reference 6). For these relatively thin divider plate assemblies, the cracks due to PWSCC tend to be very shallow (approximately 0.08 inches) and have not grown in depth since detection. These cracks are located in steam generators whose divider plates were provided primarily by one manufacturer. In addition, these cracks are believed to have initiated as a result of significant cold work introduced through surface grinding and stub runner distortion primarily attributed to hydrostatic testing of the steam generators. Analyses by the industry in the foreign country further indicated that distortion of the stub runner is only expected to occur in thinner divider plates (i.e., 1.3 inches thick or less). The cracks are primarily oriented parallel to the divider-plate-to-stub-runner weld and are located in the stub runner.

The one instance of cracking in a divider plate assembly with a thickness greater than 1.3 inches occurred in a divider plate with a thickness of approximately 2.4 inches (References 2 and 6). The cracking in this case occurred near manufacturing marks on the upper end of the stub runner used for locating tubesheet holes. These cracks were also estimated to be approximately 0.08 inches deep. This new operating experience indicates that the cracks due to PWSCC in the divider plate assemblies remain relatively shallow and do not appear to propagate after initial detection.

The foreign operating experience also indicates that fabrication issues (e.g., a misalignment between the stub runner plate and the divider plate after welding and subsequent realignment) may cause additional residual stresses and strains. Such residual stresses and strains resulting from a special plant-specific circumstance can promote PWSCC in the divider plate assemblies, although generally the residual stresses and strains are relaxed following crack growth.

The U.S. industry has performed analyses assuming a fully degraded divider plate assembly (References 6-9). For structural analyses, a fully degraded divider plate was one containing a through-wall crack extending the entire length of the component. For assessing the significance of flow bypassing the tubes, a fully degraded divider plate assumed a through-wall crack that spanned the entire length of the divider plate with a crack opening area determined from the most-limiting design basis accident. These industry analyses concluded the following regarding a fully degraded divider plate:

(a) It does not affect the design function and technical justification of the tubesheet, channel head, divider plate, tube-to-tubesheet weld, and lower shell;

(b) It does not affect the design function and technical justification of the significant tubesheet junctions and welds;

(c) It does not result in any changes to the conclusions or the level of conservatism for the analyses used to support tube plugs, sleeves, and alternate repair criteria;

(d) It does not significantly affect the heat transfer during normal operation (high power and forced flow), has a negligible impact for natural circulation conditions during design basis accidents, and does not adversely impact post loss-of-coolant accident (LOCA) long-term cooling analyses; and

(e) It is more limiting than the intact divider plate in performing design basis structural analyses of the divider plate during a LOCA.

The industry analyses (Reference 4) also indicate that cracks due to PWSCC in the divider plate assemblies are highly unlikely to affect the integrity of other pressure boundary components (i.e., the channel head and tube-to-tubesheet welds). These analyses are not considered time-limited aging analyses, but were performed to assess the possible implications of potential cracks in the divider plate assembly. For the channel head, a crack was assumed to exist at the junction of the divider plate and the channel head. Assuming this flaw would grow by fatigue for 40 years (which ignores the time for crack initiation in the divider plate), the integrity of the channel head was not compromised. In fact, there appears to be significant margin such that even if it were assumed to grow for 60 years, it would not compromise the channel head integrity. This crack growth analysis ignores the time for crack initiation which operating experience suggests is approximately 14-23 years (Reference 2).

For the tube-to-tubesheet welds, the cracks would have to turn upward and grow into the tubesheet cladding and then propagate into the welds. In addition, for some steam generators, the chromium content of the tubesheet cladding and tube-to-tubesheet welds is high enough to provide sufficient resistance to PWSCC.

The staff notes that the inservice inspections performed in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) include periodic inspections of steam generator head welds and tubesheet-to-head welds. The inspections are performed using volumetric examination (ASME Code Section XI, Item Numbers B2.30 and B2.40 in Table IWB-2500-1). The examination can confirm the structural integrity of the steam generator head welds and tubesheet-to-head welds and may provide an opportunity to detect any unexpected degradation (e.g., through observing leakage). Although these inspections are not specifically targeted to the divider plate assemblies or the region where the divider plate intersects with the tubesheet and channel head, these inspections provide an opportunity to visually detect unexpected degradation.

In addition, utilities in the U.S. perform various visual inspections of the channel head internal area when the steam generators are accessed for steam generator tube inspections. These visual inspections provide an opportunity to identify cracking if it were to grow into the pressure boundary components through identification of rust stains and gross cracking or distortion of the divider plate assembly. GALL Report, Revision 2, AMP XI.M19 does not currently include these inspections in the program scope. This LR-ISG revises AMP XI.M19 to include such visual inspections to manage cracking due to PWSCC of divider plate assemblies. As a result, this LR-ISG identifies the primary water chemistry program and steam generator program as existing AMPs that manage the aging of divider plate assemblies.

In summary, operating experience indicates there is a low likelihood of occurrence of cracking in the divider plate assembly, and when cracking occurs, there is a general lack of progression. Industry analyses indicate that there are no structural integrity concerns associated with the cracking. In addition, there are no adverse effects on other analyses (e.g., tube repair criteria, tube repair methods, design basis analyses) if cracking were to occur. Therefore, the staff concludes the following related to aging management for divider plate assemblies:

- For units with divider plate assemblies fabricated with Alloy 690 and Alloy 690 weld materials, a plant-specific AMP is not necessary given their low susceptibility to PWSCC.

- For units with divider plate assemblies fabricated with Alloy 600 or Alloy 600 weld materials, if the analyses performed by the industry (Reference 4) are applicable and bounding for the unit, the primary water chemistry program is supplemented with a general visual inspection of the steam generator channel head (as part of the steam generator program as discussed in this LR-ISG). The purpose of the visual inspection is to identify rust stains or other abnormal conditions which could indicate the presence of cracking (e.g., distortion of divider plates). The general visual inspection is performed on each steam generator at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections.

- For units with divider plate assemblies fabricated with Alloy 600 or Alloy 600 weld materials, if the industry analyses are not bounding for their unit, a plant-specific AMP is necessary or a rationale is necessary for why such a program is not needed. A plant-specific AMP (one beyond the primary water chemistry program and the steam generator program including channel head visual inspections) may include a one-time inspection that is capable of detecting cracking to verify the effectiveness of the existing programs.

Evaluation of PWSCC in Tube-to-Tubesheet Welds

In determining the AMPs needed to manage the effects of PWSCC for the tube-to-tubesheet welds, the staff considered the susceptibility of the weld material to PWSCC and the safety significance of any PWSCC should it occur in the welds.

The susceptibility of a material, including the tube-to-tubesheet weld material, to stress corrosion cracking depends on three main factors: susceptible material, conducive environment, and sufficiently high tensile stress. For cracking to occur, specific combinations of those three factors are needed. Elimination of any one of the factors effectively precludes stress corrosion cracking. The operating environment (primarily water chemistry and temperature) in pressurized water reactors is considered conducive to the occurrence of stress corrosion cracking. Therefore, the staff considered the materials and stresses in its assessment of the susceptibility of tube-to-tubesheet welds and tubesheet cladding to stress corrosion cracking.

Nickel alloys are potentially susceptible to PWSCC (i.e., stress corrosion cracking in reactor coolant) as evidenced by operating experience and research. In general, the higher the chromium content in a nickel alloy, the more resistant the alloy is to PWSCC. However, it is difficult to specify a certain chromium content below which the material would be susceptible to PWSCC because PWSCC is a complicated phenomenon involving many factors. For example, even with the same chromium content, materials may have different susceptibilities to PWSCC based on the heat treatment of the material. Thermally-treated Alloy 600 is more resistant to PWSCC than mill-annealed Alloy 600 even though the overall chromium contents are the same. Although research provides some insights on the initiation and growth of PWSCC, it is difficult to extrapolate these results to predict material behavior in operating plants given the complex interactions and effects of multiple parameters (e.g., aggressive chemical species under a wide range of operating and residual stresses). In light of the discussion above, operating experience is a very useful predictor of material performance since it provides a direct indication of how a material will perform in the actual operating environment with representative applied and residual stresses.

In general, operating experience with the primary coolant system has shown a marked increase in the resistance of nickel alloys to PWSCC when the chromium content exceeds approximately 18 percent since little, or no, cracking has been observed to date in materials with chromium contents in the range of 18 to 22 percent (e.g., Alloy 82). In addition, the staff is not aware of any PWSCC in nickel alloys with chromium contents greater than 22 percent in nuclear power plant applications, which includes Alloy 690 type materials (approximately 27 to 31 percent chromium).

The chromium content of the tube-to-tubesheet autogenous welds depends on both the tube material (Alloy 600 or Alloy 690) and the tubesheet cladding material (Alloy 600 type material such as Alloy 82/182, or Alloy 690 type material such as Alloy 52/152). These variations in chromium content result in four potential combinations of tube material and tubesheet cladding material. For tube-to-tubesheet welds made between Alloy 600 tubes and Alloy 600 type cladding, the welds would normally be considered highly susceptible to PWSCC given the low chromium content of the welds. For tube-to-tubesheet welds made between Alloy 690 tubes and Alloy 690 type cladding, the welds would be highly resistant to PWSCC given the high chromium content of the welds. The staff is not aware of tube-to-tubesheet welds that would have been made with Alloy 600 tubes and Alloy 690 cladding so this material combination was not evaluated. For tube-to-tubesheet welds made between Alloy 690 tubes and Alloy 600 type cladding, a more detailed assessment is needed since the chromium content could potentially be low enough to make the welds more susceptible to PWSCC.

If a steam generator tubesheet has Alloy 600 type cladding material, it would normally have both Alloy 82 (18 to 22 percent chromium) and Alloy 182 (13 to 17 percent chromium) cladding material with the majority of the tubesheet surface being clad with Alloy 82. Regions clad with Alloy 182 would normally include the central region of the tubesheet, a strip adjacent to the divider plate (potentially involving the first row of tubes), and local repairs. As a result, for tube-to-tubesheet welds made between Alloy 690 tubes and Alloy 600 type cladding material, the chromium content of the welds will vary depending on whether the welds were made with Alloy 82 or Alloy 182 cladding. The industry analyses (Reference 4) indicate that the weld chromium content for Alloy 690 tubes and Alloy 82 cladding can range from approximately 24 to 26 percent chromium and the weld chromium content for Alloy 690 tubes and Alloy 182 cladding can range from approximately 21 to 23 percent. Therefore, the tube-to-tubesheet welds made with Alloy 82 cladding would be considered fairly resistant to PWSCC, and the tube-to-tubesheet welds made with Alloy 182 cladding would be considered less resistant to PWSCC than the tube-to-tubesheet welds formed from Alloy 82 cladding material.

To initiate a stress corrosion crack, tensile stresses are required. With respect to the stresses, the industry analyses indicated that the operating stresses on the primary face (lower surface) of the tubesheet are compressive except for a region adjacent to the divider plate (tubesheet pass partition lane). However, the staff noted that these analyses did not consider weld residual stresses in the tubesheet cladding. Therefore, the staff found that the possibility of PWSCC in the tubesheet cladding cannot be excluded solely based on these stress analyses.

In reviewing operating experience, the staff has not identified any instances where cracks have been reported in the tubesheet cladding. Although it is unlikely that any inspections looking specifically for cracking have been performed, if cracking were prevalent, it would have most likely been detected during the performance of steam generator tube inspections. During outages in which tube inspections are performed, tube plugs are visually inspected and cameras are used to guide probes through the tubes. If cracking were to occur in the cladding, it would potentially expose the carbon steel tubesheet base material. If this were to occur, corrosion of the carbon steel is possible. This corrosion would lead to rust stains on the cladding which can be readily detected.

Except for one instance caused by a maintenance activity (Reference 5), the staff is not aware of any utility in the U.S. observing degradation of the tubesheet cladding as would be evidenced by the presence of rust stains. In addition, if corrosion of the carbon or low alloy steel were to occur, this could potentially lead to deformation of the tubes near the tube ends since the corrosion products associated with steel corrosion occupy more volume than the original material. The staff is not aware of any utility identifying denting (inward deformation) of a tube near the tube end as a result of corrosion of the tubesheet. This operating experience provides additional confidence that the weld chemistry analysis of the tubesheet cladding performed by the industry is reasonable. In addition, if a crack due to PWSCC in the tube-to-tubesheet weld propagates into the tube, periodic steam generator tube inspections will detect the crack propagation.

The tube-to-tubesheet joint serves a reactor coolant pressure boundary function. This joint is analyzed to ensure it will hold the tube in place during all normal operating, transient, and accident loading conditions with margin. The tube-to-tubesheet joint consists of a tube-to-tubesheet weld and an interference fit between the tube and the tubesheet (since the tube is expanded against a hole in the tubesheet for that tube). Most steam generators were originally designed such that the tube-to-tubesheet welds were structural welds taking no credit for the ability of tube expansion against the tubesheet bore (i.e., the interference fit) to resist axial loads on the tube. As a result, the tube-to-tubesheet welds are classified as structural welds.

As degradation was observed in the portion of the tube in the tubesheet, licensees proposed and NRC approved, a redefinition of the reactor coolant pressure boundary such that, in these cases, the welds no longer serve a reactor coolant pressure boundary function. In these cases, it was shown that only a portion of the interference fit between the tube and the tubesheet was sufficient to hold the tube in place during all normal operating, transient, and accident loading conditions with margin, and that primary-to-secondary leakage through this joint could be limited to values assumed in the accident analyses (e.g., through testing, analyses, or monitoring operational leakage). If the tube-to-tubesheet weld does not serve a reactor coolant pressure boundary/safety function, a plant-specific AMP for these welds is not needed. Even for units where the tube-to-tubesheet weld is considered as reactor coolant pressure boundary, the interference fit between the tube and the tubesheet will provide some resistance to the axial loads acting on the tube and will limit the amount of primary-to-secondary leakage, thereby reducing the safety significance of any cracking in the welds should it occur.

As previously discussed, utilities in the U.S. perform various visual inspections of the channel head internal area when the steam generators are accessed for steam generator tube inspections. These visual inspections provide an opportunity to identify cracking if it were to grow into the pressure boundary components through identification of rust stains or other abnormal observations. GALL Report, Revision 2, AMP XI.M19 does not currently include these inspections in the program scope. This LR-ISG revises GALL Report AMP XI.M19 to include such visual inspections to manage cracking due to PWSCC of tube-to-tubesheet welds. Therefore, this LR-ISG identifies the primary water chemistry program and steam generator program as existing AMPs that manage cracking due to PWSCC for the tube-to-tubesheet welds.

As a result of the above (operating experience with Alloy 600 and Alloy 690 and their weld materials, the chromium content of the tube-to-tubesheet welds, the visual inspection of the tubesheet region performed during tube inspection outages, the operating experience with the performance of the tubesheet cladding, and the safety significance of the welds), the staff concludes the following:

- For units with Alloy 600 steam generator tubes and for which an alternate repair criteria such as C*, F*, W*, or H* has been permanently approved for both the hot- and cold-leg side of the steam generator, the weld is no longer part of the reactor coolant pressure boundary and a plant-specific AMP is not necessary;

- For units with Alloy 600 steam generator tubes and for which there is no permanently approved alternate repair criteria such as C*, F*, W*, or H* or where permanent approval only applies to the hot- or cold-leg side of the steam generator, a plant-specific AMP is necessary;

- For units with thermally treated Alloy 690 steam generator tubes and with tubesheet cladding using Alloy 690 weld material, a plant-specific AMP is not necessary;

- For units with thermally treated Alloy 690 steam generator tubes and with tubesheet cladding using Alloy 600 weld material, a plant-specific AMP is necessary unless the applicant confirms that the industry's analyses for tube-to-tubesheet weld cracking (e.g., chromium content for the tube-to-tubesheet welds is approximately 22 percent and the tubesheet cladding is in compression) are applicable and bounding for its unit, and the applicant will perform general visual inspections of the tubesheet region looking for evidence of cracking (e.g., rust stains on the tubesheet cladding) as part of the steam generator program. In lieu of a plant-specific AMP, the applicant may provide a rationale for why a plant-specific AMP is not necessary.

Evaluation of Steam Generator Head and Tubesheet Degradation

Foreign and domestic operating experience indicates that loss of material due to boric acid corrosion can occur in the steel base material of the steam generator channel head and tubesheet. This operating experience is discussed in NRC Information Notice 2013-20, "Steam Generator Channel Head and Tubesheet Degradation" (Reference 5). This corrosion primarily occurs when the plant is shut down and the steam generator channel head internal area is exposed to oxygen. This operating experience indicates that, if the steam generator cladding is compromised (e.g., due to cracking, manufacturing defects, maintenance, or foreign material impingement damage), loss of material due to boric acid corrosion could occur in the steel base material of the channel head and tubesheet.

Given the above operating experience, the staff concludes that aging management is needed to address this potential degradation. One means to effectively manage this aging effect is to control the reactor water chemistry to mitigate the loss of material due to boric acid corrosion for the base material in the event that the cladding is compromised and to perform periodic visual inspections of the cladded surfaces within the steam generator to detect anomalous conditions (e.g., rust stains).

As a result, this LR-ISG revises GALL Report AMP XI.M19 to include steam generator primary side internal surfaces and to indicate that visual inspections of these surfaces should be performed at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections.

Description of the Major Changes to Aging Management Guidance

Based on the information discussed above, the staff concluded that changes to the aging management guidance in the GALL Report and the SRP-LR are warranted.

Summary of SRP-LR Changes:

Revision 2 of the SRP-LR is revised to reflect that aging of the steam generator divider plate assembly is managed through both the water chemistry and steam generator programs. In addition, it is revised to indicate that a plant-specific AMP is not necessary for a unit with divider plate assemblies fabricated of Alloy 600 or Alloy 600 weld materials if the analyses performed by the industry (Reference 4) are applicable and bounding for the unit.

Similarly, the SRP-LR is revised to reflect that aging of the tube-to-tubesheet welds is managed through both the water chemistry and steam generator programs. In addition, it is revised to indicate that a plant-specific AMP is not necessary for a unit with thermally treated Alloy 690 steam generator tubes and tubesheet cladding using Alloy 600 type material, if the analyses performed by the industry (Reference 4) for tube-to-tubesheet weld cracking (e.g., chromium content for the tube-to-tubesheet welds is approximately 22 percent and that the tubesheet cladding is in compression) are applicable and bounding for the unit. The FSAR supplement for GALL Report AMP XI.M19 (SRP-LR Table 3.0-1) is also updated.

The revision to the SRP-LR is included in Appendix A of this LR-ISG.

Summary of GALL Report Changes:

Section XI.M19 “Steam Generators” of the GALL Report, Revision 2 is revised to reflect that aging management of the channel head, divider plate assembly, tubesheet, and tube-to-tubesheet welds is within the scope of the program. Associated AMR items in the GALL Report are also modified to add AMP XI.M19 as an existing program that manages cracking due to PWSCC for the divider plate assembly and tube-to-tubesheet welds and loss of material due to boric acid corrosion for steam generator head internal surfaces, including the tubesheet primary side. In addition, changes are made to update references in the AMP and to further clarify program elements related to the applicant’s Quality Assurance (QA) program.

The revision to the GALL Report is included in Appendix B of this LR-ISG.

**ACTION**

Applicants should use Appendices A and B in preparing their LRAs to be consistent with the GALL Report and this LR-ISG.

**NEWLY IDENTIFIED SYSTEMS, STRUCTURES, AND COMPONENTS UNDER 10 CFR 54.37(b)**

The NRC is not proposing to treat the revised recommendations for managing aging effects associated with steam generator divider plate assemblies, tube-to-tubesheet welds, heads, and tubesheets as “newly identified” systems, structures, components (SSCs) under 10 CFR 54.37(b). Therefore, any additional action for such SSCs, which the NRC may impose upon current holders of renewed operating licenses under 10 CFR Part 54, would not fall within the scope of 10 CFR 54.37(b). The NRC would address compliance with the requirements of 10 CFR 50.109, “Backfitting,” before imposing any new aging management requirements on current holders of renewed operating licenses (see discussion below).

**BACKFITTING**

This LR-ISG contains guidance on one acceptable approach for managing the associated aging effects occurring during the period of extended operation for steam generator divider plate assemblies, tube-to-tubesheet welds, heads, and tubesheets. The staff intends to use the guidance in this LR-ISG when reviewing current and future license renewal applications. The staff also intends to use the LR-ISG in evaluating voluntary, licensee-initiated changes to previously-approved AMPs.  Existing holders of renewed operating licenses may follow the guidance in this LR-ISG, but would not be required to do so.

Backfitting

Issuance of this LR-ISG does not constitute backfitting as defined in the Backfit Rule for nuclear power plants, 10 CFR 50.109(a)(1), and the staff did not prepare a backfit analysis for issuing this LR-ISG.  There are several rationales for this conclusion, depending on the status of the nuclear power plant licensee under 10 CFR Parts 50 and 54.

*Licensees currently in the license renewal process* - The backfitting provisions in 10 CFR 50.109 are not applicable to an applicant for a renewed license.  Therefore, issuance of this LR-ISG would not constitute backfitting as defined in 10 CFR 50.109(a)(1).

*Licensees that already hold a renewed license* - This guidance would be nonbinding and the LR-ISG would not require current holders of renewed licenses to take any action (i.e., programmatic or plant hardware changes for managing the associated aging effects for components within the scope of this LR-ISG).  The current holders of renewed licenses could treat the information presented in this LR-ISG as “operating experience” information and consider this information to ensure that relevant AMPs are, and will remain, effective.  If, in the future, the NRC decides to take additional action and impose requirements for managing the associated aging effects for components within the scope of this LR-ISG, then the NRC would follow the requirements of the Backfit Rule.

*Current* 10 CFR *Part 50 operating license holders that have not yet applied for renewed licenses* - The backfitting provisions in 10 CFR 50.109 do not apply to any future applicant for license renewal.  Therefore, issuance of this LR-ISG would not constitute backfitting as defined in 10 CFR 50.109(a)(1).

**CONGRESSIONAL REVIEW ACT**

This LR-ISG is a rule as defined in the Congressional Review Act (5 U.S.C. §§ 801-808).  The Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

**APPENDICES**

Appendix A describes the changes (redline/strikeout) to the guidance in NUREG-1800 (SRP-LR), Revision 2.

Appendix B describes the changes (redline/strikeout) to the guidance in NUREG-1801 (GALL Report), Revision 2.

Appendix C describes the resolution of public comments.

Appendix D provides the final revised guidance in NUREG-1800 (SRP-LR), Revision 2.

Appendix E provides the final revised guidance in NUREG-1801 (GALL Report), Revision 2.

## REFERENCES


1. "Learnings from Investigations on SG [Steam Generator] Divider Plates: Coupling Field Characterizations with Numerical Mechanical Simulation," F. Rossillon et al., Nuclear Engineering and Design, Vol. 269 (Special volume for SMiRT21), pp. 45-51, April 2014.

2. "Learnings from EdF Investigations on SG [Steam Generator] Divider Plates and Vessel Head Nozzles. Evidence of Prior Deformation Effect on Stress Corrosion Cracking," D. DeForge et al., Fontevraud 7, September 26-30, 2010.

3. "Destructive Examinations on Divider Plates from Decommissioned Steam Generators Affected by Superficial Stress Corrosion Cracks," S. Miloudi et al., 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, 2012.

4. EPRI 3002002850, "Steam Generator Management Program: Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly," Electric Power Research Institute, Palo Alto, CA, October 2014.

5. NRC Information Notice 2013-20, "Steam Generator Channel Head and Tubesheet Degradation," U.S. Nuclear Regulatory Commission, October 3, 2013.

6. EPRI Presentation, "NRC/Industry Meeting Regarding Tube-to-Tubesheet Weld and Divider Plate Cracking Report," July 30, 2015 (ADAMS Accession Number ML15211A507).

7. EPRI 1020988, "Steam Generator Management Program: Phase II Divider Plate Cracking Engineering Study," Electric Power Research Institute, Palo Alto, CA, November 2010.

8. EPRI 1016552, "Divider Plate Cracking in Steam Generators: Results of Phase II: Evaluation of the Impact of a Cracked Divider Plate on LOCA and Non-LOCA Analyses," Electric Power Research Institute, Palo Alto, CA, November 2008.

9. EPRI 1014982, "Divider Plate Cracking in Steam Generators - Results of Phase 1: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material," Electric Power Research Institute, Palo Alto, CA, June 2007.

**Appendix A: Changes to the Guidance in NUREG-1800 (SRP-LR), Revision 2**

In this appendix, the strikeout indicate where deletion is made and the underscore indicates where addition is made.

**Revised Section 3.1.2.2.11.1 (Acceptance Criteria for Divider Plate Assemblies)**

Foreign operating experience in steam generators with a ~~similar~~ design similar to that of Westinghouse steam generators (particularly Model 51) has identified ~~extensive~~ crack~~ing~~s due to primary water stress corrosion cracking (PWSCC) in steam generator (SG) divider plate assemblies fabricated of Alloy 600 and/or the associated Alloy 600 weld materials, even with proper primary water chemistry ~~(EPRI TR-1014982)~~. Cracks have been detected in the stub runner with depths typically about 0.08 inches (EPRI 3002002850). ~~, adjacent to the tubesheet/stub runner weld and with depths of almost a third of the divider plate thickness. Therefore, the water chemistry program may not be effective in managing the aging effect of cracking due to PWSCC in SG divider plate assemblies. This is of particular concern for steam generators where the tube-tubesheet welds are considered structural welds and/or where the divider plate assembly contributes to the mechanical integrity of the tubesheet.~~

~~Although these SG divider plate cracks may not have a significant safety impact in and of themselves, these cracks could impact adjacent items, such as the tubesheet and the channel head, if they propagate to the boundary with these items. For the tubesheet, PWSCC cracks in the divider plate could propagate to the tubesheet cladding with possible consequences to the integrity of the tube/tubesheet welds. For the channel head, the PWSCC cracks in the divider plate could propagate to the SG triple point and potentially affect the pressure boundary of the SG channel head.~~

All but one of these instances of cracking have been detected in divider plate assemblies that are approximately 1.3 inches in thickness. For the cracks in the 1.3-inch thick divider plate assemblies, the cracks tend to be parallel to the divider-plate-to-stub-runner weld (i.e., run horizontally in parallel to the lower surface of the tubesheet). For the one instance of cracking in a divider plate assembly with a thickness greater than 1.3 inches, the cracking occurred in a divider plate assembly with a thickness of approximately 2.4 inches near manufacturing marks on the upper end of the stub runner used for locating tubesheet holes. These flaws were estimated to be approximately 0.08-inch deep.

Although these instances indicate that the water chemistry program may not be sufficient to manage cracking due to PWSCC in SG divider plate assemblies, analyses by the industry indicate that PWSCC in the divider plate assembly does not pose a structural integrity concern for other steam generator components (e.g., tubesheet and tube-to-tubesheet welds) and does not adversely affect other safety analyses (e.g., analyses supporting tube plugging and repairs, tube repair criteria, and design basis accidents). In addition, the industry analyses indicate that flaws in the divider plate assembly will not adversely affect the heat transfer function (as a result of bypass flow) during normal forced flow operation, during natural circulation conditions (assessed in the analyses of various design basis accidents), or in the event of a loss-of-coolant accident (LOCA).

Furthermore, additional industry analyses indicate that PWSCC in the divider plate assembly is unlikely to adversely impact adjacent items, such as the tubesheet cladding, tube-to-tubesheet welds, and channel head. Therefore,

- For units with divider plate assemblies fabricated of Alloy 690 and Alloy 690 type weld materials, a plant-specific aging management program (AMP) is not necessary.

- For units with divider plate assemblies fabricated of Alloy 600 or Alloy 600 type weld materials, if the analyses performed by the industry (EPRI 3002002850) are applicable and bounding for the unit, a plant-specific AMP is not necessary.

- For units with divider plate assemblies fabricated of Alloy 600 or Alloy 600 type weld materials, if the industry analyses (EPRI 3002002850) are not bounding for the applicant's unit, a plant-specific AMP is necessary or a rationale is necessary for why such a program is not needed. A plant-specific AMP (one beyond the primary water chemistry and the steam generator programs) may include a one-time inspection that is capable of detecting cracking to verify the effectiveness of the water chemistry and steam generator programs and the absence of PWSCC in the divider plate assemblies.

~~The existing program relies on control of reactor water chemistry to mitigate cracking due to PWSCC. The GALL Report recommends that a plant-specific AMP be evaluated, along with the primary water chemistry program, because the existing primary water chemistry program may not be capable of mitigating cracking due to PWSCC.~~ The existing programs rely on control of reactor water chemistry to mitigate cracking due to PWSCC and general visual inspections of the channel head interior surfaces (included as part of the steam generator program). The GALL Report recommends further evaluation for a plant-specific AMP to confirm the effectiveness of the primary water chemistry and steam generator programs as described in this section. Acceptance criteria for a plant-specific AMP are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR). In place of a plant-specific AMP, the applicant may provide a rationale to justify why a plant-specific AMP is not necessary.

**Revised Section 3.1.3.2.11.1 (Review Procedures for Divider Plate Assemblies)**

The GALL Report recommends ~~that~~further evaluation of a plant-specific AMP ~~be evaluated~~, along with the primary water chemistry and steam generator programs, to manage cracking due to PWSCC in nickel alloy divider plate assemblies. ~~made of Alloy 600 and/or the associated Alloy 600 weld materials for steam generators with a similar design to that of Westinghouse Model 51. The effectiveness of the chemistry control program should be verified to ensure that cracking due to PWSCC is not occurring. The reviewer verifies the materials of construction of the applicant's SG divider plate assembly. If these materials are susceptible to cracking, the reviewer verifies that the applicant has evaluated the potential for cracking in the divider plate to~~

~~propagate into other components (e.g., tubesheet cladding). If propagation into these other components is possible, the reviewer verifies if the applicant has described an inspection program (examination technique and frequency) for ensuring that no cracks are propagating into other items (e.g., tube sheet and channel head) that could challenge the integrity of those items. The reviewer reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of this aging effect.~~For divider plate assemblies fabricated of Alloy 690 and Alloy 690 type welding materials, a plant-specific AMP is not necessary. For divider plate assemblies made of Alloy 600 or Alloy 600 type welding materials, the reviewer verifies that the applicant has an adequate basis for concluding that the analyses performed by the industry (EPRI 3002002850) assessing the significance of divider plate cracking are applicable and bounding for the conditions at its unit. If the industry's analyses are not bounding, the reviewer evaluates the applicant's plant-specific AMP on a case-by-case basis to ensure that an adequate plant-specific AMP will be in place for the management of this aging effect or the reviewer reviews the applicant's rationale (e.g., a more detailed plant-specific evaluation than performed by the industry) for why such a plant-specific AMP is not necessary.

**Revised Section 3.1.2.2.11.2 (Acceptance Criteria for Tube-to-Tubesheet Welds)**

Cracking due to PWSCC could occur in steam generator nickel alloy tube-to-tubesheet welds exposed to reactor coolant. ~~Unless the NRC has approved a redefinition of the pressure boundary in which the tube-to-tubesheet weld is no longer included, the effectiveness of the primary water chemistry program should be verified to ensure cracking is not occurring:~~
The acceptance criteria for this review are:

- For ~~plants~~units with Alloy 600 steam generator tubes ~~that have not been thermally treated~~ and for which an alternate repair criteria such as C*, F*, ~~or~~ W*, or H* has been permanently approved for both the hot- and cold-leg side of the steam generator, the weld is no longer part of the reactor coolant pressure boundary and ~~no~~a plant-specific AMP ~~aging management program~~ is not necessary ~~required~~;

- For ~~plants~~units with Alloy 600 steam generator tubes ~~that have not been thermally treated~~, if ~~and for which~~ there is no permanently approved alternate repair criteria such as C*, F*, ~~or~~ W*, or H* or permanent approval applies to only either the hot- or cold-leg side of the steam generator, a plant-specific AMP is necessary ~~required~~;

- ~~For plants with Alloy 600TT steam generator tubes and for which an alternate repair criteria such as H* has been permanently approved, the weld is no longer part of the pressure boundary and no plant specific aging management program is required;~~

- ~~For plants with Alloy 600TT steam generator tubes and for which there is no alternate repair criteria such as H* permanently approved, a plant-specific AMP is required;~~

- For ~~plants~~ units with thermally treated Alloy 690~~TT~~ steam generator tubes and with ~~Alloy 690~~ tubesheet cladding using Alloy 690 type material, a plant-specific AMP is not necessary ~~the water chemistry is sufficient, and no further action or plant-specific aging management program is required~~;

- For ~~plants~~ units with thermally treated Alloy 690~~TT~~ steam generator tubes and with tubesheet cladding using Alloy 600 ~~tubesheet cladding~~ type material, ~~either a plant-specific program or a rationale for why such a program is not needed is required~~ a plant-specific AMP is necessary unless the applicant confirms that the industry's analyses for tube-to-tubesheet weld cracking (e.g., chromium content for the tube-to-tubesheet welds is approximately 22 percent and the tubesheet primary face is in compression as discussed in EPRI 3002002850) are applicable and bounding for the unit, and the applicant will perform general visual inspections of the tubesheet region looking for evidence of cracking (e.g., rust stains on the tubesheet cladding) as part of the steam generator program. In lieu of a plant-specific AMP, the applicant may provide a rationale for why a plant-specific AMP is not necessary.

The existing programs rel~~ies~~y on control of reactor water chemistry to mitigate cracking due to PWSCC and visual inspections of the steam generator head interior surfaces. Along with the primary water chemistry and steam generator programs, ~~T~~the GALL Report recommends ~~that~~ further evaluation of a plant-specific AMP ~~be evaluated, along with the primary water chemistry program~~ to confirm the effectiveness of the primary water chemistry and steam generator programs in certain circumstances.~~, because the existing primary water chemistry program may not be capable of mitigating cracking due to PWSCC.~~ A plant-specific AMP may include a one-time inspection that is capable of detecting cracking to confirm the absence of PWSCC in the tube-to-tubesheet welds. Acceptance criteria for a plant-specific AMP are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR). In place of a plant-specific AMP, the applicant may provide a rationale to justify why a plant-specific AMP is not necessary.

**Revised Section 3.1.3.2.11.2 (Review Procedures for Tube-to-Tubesheet Welds)**

The GALL Report recommends ~~that~~ further evaluation for a plant-specific AMP ~~be evaluated~~, along with the primary water chemistry and steam generator programs, to manage cracking due to PWSCC in nickel alloy ~~recirculating~~ steam generator ~~nickel alloy~~ tube-to-tubesheet welds exposed to reactor coolant. ~~The effectiveness of the primary water chemistry program should be verified to ensure that cracking due to PWSCC is not occurring~~. The reviewer verifies the combination of materials of construction of the steam generator tubes and tubesheet cladding and the classification of the tube-to-tubesheet weld (i.e., whether it is part of the reactor coolant pressure boundary). If th~~is~~e combination and classification require~~s further evaluation~~ a plant-specific AMP, the reviewer reviews the applicant's proposed program on a case-by-case basis to ensure adequate management of this aging effect ~~that an adequate program will be in place for the management of this aging effect~~. Alternatively, the reviewer evaluates applicant's rationale for why a plant-specific AMP is not necessary.

**Changes to Section 3.1.6**

Section 3.1.6 is revised to add the following references:

31. EPRI 3002002850, "Steam Generator Management Program: Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly," Electric Power Research Institute, Palo Alto, CA, October 2014.

32. EPRI 1020988, "Steam Generator Management Program: Phase II Divider Plate Cracking Engineering Study," Electric Power Research Institute, Palo Alto, CA, November 2010.

33. EPRI 1016552, "Divider Plate Cracking in Steam Generators: Results of Phase II: Evaluation of the Impact of a Cracked Divider Plate on LOCA and Non-LOCA Analyses," Electric Power Research Institute, Palo Alto, CA, November 2008.

34. EPRI Presentation, "NRC/Industry Meeting Regarding Tube-to-Tubesheet Weld and Divider Plate Cracking Report," July 30, 2015.

**Changes to SRP-LR Table 3.1-1 (AMR Items)**

| Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report | | | | | | | |
|---|---|---|---|---|---|---|---|
| **ID** | **Type** | **Component** | **Aging Effect/Mechanism** | **Aging Management Programs** | **Further Evaluation Recommended** | **Rev2 Item** | **Rev1 Item** |
| 25 | PWR | Steel (with nickel-alloy cladding) or nickel alloy steam generator primary side components: divider plate and tube-to-tube sheet welds exposed to reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" | Yes, plant-specific (See subsections 3.1.2.2.11.1 and 3.1.2.2.11.2) | IV.D1.RP-367<br>IV.D1.RP-385<br>IV.D2.RP-185 | IV.D1-6(RP-21)<br>N/A<br>IV.D2-4(R-35) |
| 127a | PWR | Steel (with stainless steel or nickel alloy cladding) steam generator heads and tubesheets exposed to reactor coolant | Loss of material due to boric acid corrosion | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" | No | IV.D1.R-436a<br>IV.D2.R-440a | N/A |

**Changes to SRP-LR Table 3.0-1 (FSAR supplement)**

As addressed in LR-ISG-2011-02, “Aging Management Program for Steam Generators,” GALL Report AMP XI.M19 relies on the guidance in Revision 3 of Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines.” This change with updated references to industry steam generator guidelines is reflected in the following table on the FSAR supplement for GALL AMP XI.M19 and in Appendix B of this LR-ISG.

**Table 3.0-1 FSAR Supplement for Aging Management of Applicable Systems**

| GALL Chapter | GALL Program | Description of Program | Implementation Schedule* | Applicable GALL Report and SRP-LR Chapter References |
|---|---|---|---|---|
| XI.M19 | Steam Generators | This program manages the aging of steam generator tubes, plugs, sleeves, divider plate assemblies (as applicable), tube-to-tubesheet welds, heads (interior surfaces of channel or lower/upper heads), tubesheets (primary side), and secondary side components that are contained within the steam generator. This program consists of aging management activities for the steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator in accordance with the plant technical specifications and includes commitments to NEI 97-06, Revision 3 and the associated EPRI guidelines. This program also performs general visual inspections of the steam generator heads (internal surfaces) looking for evidence of cracking or loss of material (e.g., rust stains) at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. The program includes foreign material exclusion as a means to inhibit wear degradation, and secondary side maintenance activities, such as sludge lancing, for removing deposits that may contribute to component degradation. The program performs volumetric examination on steam generator tubes in accordance with the requirements in the technical specifications to detect aging effects, if they should occur. The technical specifications require condition monitoring (explicitly) and operational assessments (implicitly) to be performed to ensure that the tube integrity will be maintained until the next inspection. | Existing program | GALL IV / SRP 3.1 |

## Appendix B: Changes to the Guidance in NUREG-1801 (GALL Report), Revision 2

In this appendix, the strikeout indicates where deletions are made and the underscore indicates where additions are made. The changes to GALL Report AMP XI.M19 and AMR tables are provided below. As discussed in Appendix A of this LR-ISG, the change addressed in LR-ISG-2011-02 (i.e., updated reference to Revision 3 of NEI 97-06) and updated references to the EPRI steam generator guidelines are also reflected in this appendix.

### XI.M19 STEAM GENERATORS

### Program Description

The Steam Generator program is applicable to managing the aging of steam generator tubes, plugs, sleeves, divider plate assemblies, tube-to-tubesheet welds, heads (interior surfaces of channel or lower/upper heads), tubesheet(s) (primary side), and secondary side components that are contained within the steam generator (i.e., secondary side internals). The aging of steam generator pressure vessel welds is managed by other programs such as GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry."

The establishment of a steam generator program for ensuring steam generator tube integrity is required by plant technical specifications (TSs). The steam generator tube integrity portion of the ~~technical specifications~~ TSs at each PWR contains the same fundamental requirements as outlined in the standard technical specifications of NUREG-1430, Volume 1, Rev. ~~3~~ 4, for Babcock & Wilcox (B&W) ~~pressurized water reactors (~~PWRs~~)~~; NUREG-1431, Volume 1, Rev. ~~3~~ 4, for Westinghouse PWRs; and NUREG-1432, Volume 1, Rev. ~~3~~ 4, for Combustion Engineering PWRs. The requirements pertaining to steam generators in these three versions of the standard ~~technical specifications~~ TSs are essentially identical. The ~~technical specifications~~ TSs require tube integrity to be maintained and specify performance criteria, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging or repair of flawed tubes, acceptable tube repair methods, and leakage monitoring requirements.

The nondestructive examination techniques used to inspect steam generator components covered by this program ~~tubes, plugs, sleeves, and secondary side internals~~ are intended to identify components (e.g., tubes, plugs) with degradation that may need to be removed from service (e.g., tubes), ~~or~~ repaired, or replaced, as appropriate.

The Steam Generator program at PWRs is modeled after Nuclear Energy Institute (NEI) 97-06, Revision ~~2~~ 3, "Steam Generator Program Guidelines." This program references a number of industry guidelines (e.g., the Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines, PWR Primary-to-Secondary Leak Guidelines, PWR Primary Water Chemistry Guidelines, PWR Secondary Water Chemistry Guidelines, Steam Generator Integrity Assessment Guidelines, Steam Generator In Situ Pressure Test Guidelines) and incorporates a balance of prevention, mitigation, inspection, evaluation, repair, and leakage monitoring measures. The NEI 97-06 document (a) includes performance criteria that are intended to provide assurance that tube integrity is being maintained consistent with the plant's licensing basis and (b) provides guidance for monitoring and maintaining the tubes to provide assurance that the performance criteria are met at all times between scheduled inspections of the tubes. Steam generator tube integrity can be affected by degradation of steam generator plugs, sleeves, and secondary side ~~internals~~ components. ~~Therefore, all of these components are~~

~~addressed by this aging management program (AMP).~~ The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves and secondary side ~~internals~~ components.

Degradation of divider plate assemblies, tube-to-tubesheet welds, heads (internal surfaces), or tubesheets (primary side) may have safety implications. Therefore, all of these components and the steam generator tubes, plugs, sleeves and secondary side components are addressed by this aging management program (AMP).

**Evaluation and Technical Basis**

1. ***Scope of Program:*** This program addresses degradation associated with steam generator tubes, plugs, sleeves, divider plate assemblies, tube-to-tubesheet welds, heads (interior surfaces of channel or lower/upper heads), tubesheet(s) (primary side), and secondary side components that are contained within the steam generator (i.e., secondary side internals). The program does not cover the steam generator secondary side shell, any nozzles attached to the secondary side shell or steam generator head, or the welds associated with these components. In addition, the program does not cover steam generator head welds (other than general corrosion of these welds caused as a result of degradation (defects/flaws) in the primary side cladding). ~~It does not cover degradation associated with the steam generator shell, channel head, nozzles, or welds associated with these components.~~

2. ***Preventive Actions:*** This program includes preventive and mitigative actions for addressing degradation. Preventive and mitigative measures that are part of the Steam Generator program include foreign material exclusion programs, and other primary and secondary side maintenance activities. The program includes foreign material exclusion as a means to inhibit wear degradation and secondary side maintenance activities, such as sludge lancing, for removing deposits that may contribute to degradation. Guidance on foreign material exclusion is provided in NEI 97-06. Guidance on maintenance of secondary side integrity is provided in the EPRI Steam Generator Integrity Assessment Guidelines. Primary side preventive maintenance activities include replacing plugs made with corrosion susceptible materials with more corrosion resistant materials and preventively plugging tubes susceptible to degradation.

   Extensive deposit buildup in the steam generators could affect tube integrity. The EPRI Steam Generator Integrity Assessment Guidelines, which are referenced in NEI 97-06, provide guidance on maintenance on the secondary side of the steam generator, including secondary side cleaning. Secondary side water chemistry plays an important role in controlling the introduction of impurities into the steam generator and potentially limiting their deposition on the tubes. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Water chemistry is monitored and maintained in accordance with the Water Chemistry program. The program description and evaluation and technical basis of monitoring and maintaining water chemistry are addressed in GALL AMP XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** There are currently three types of steam generator tubing used in the United States: mill annealed Alloy 600, thermally treated Alloy 600, and thermally treated Alloy 690. Mill annealed Alloy 600 steam generator tubes have experienced degradation due to corrosion (e.g., primary water stress corrosion cracking, outside diameter stress corrosion cracking, intergranular attack, pitting, and wastage) and mechanically induced phenomena (e.g., denting, wear, impingement damage, and fatigue).

Thermally treated Alloy 600 steam generator tubes have experienced degradation due to corrosion (primarily cracking) and mechanically induced phenomena (primarily wear). Thermally treated Alloy 690 tubes have only experienced tube degradation due to mechanically induced phenomena (primarily wear).

Degradation of tube plugs, sleeves, heads, tubesheet(s), and secondary side internals ha~~ve~~s also been observed, depending, in part, on the material of construction of the specific component. The potential for degradation exists for divider plate assemblies and tube-to-tubesheet welds, depending, in part, on the materials of construction for the components.

The program includes an assessment of the forms of degradation to which a component is susceptible and implementation of inspection techniques capable of detecting those forms of degradation. The parameter monitored is specific to the component and the acceptance criteria for the inspection. For example, the severity of tube degradation may be evaluated in terms of the depth of degradation or measured voltage, dependent on whether a depth-based or voltage-based tube repair criteria (acceptance criteria) is being implemented for that specific degradation mechanism. Other parameters monitored include signals of excessive deposit buildup (e.g., steam generator water level oscillations), which may result in fatigue failure of tubes or corrosion of the tubes; water chemistry parameters, which may indicate unacceptable levels of impurities; primary-to-secondary leakage, which may indicate excessive tube, plug, or sleeve degradation; and the presence of loose parts or foreign objects on the primary and secondary side of the steam generator, which may result in tube damage.

Water chemistry parameters are also monitored and controlled, as discussed in AMP XI.M2. The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines (EPRI ~~1008219~~ 1022832) provides guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines (EPRI ~~1012987~~ 3002007571) provide guidance on secondary side activities.

In summary, the NEI 97-06 program provides guidance on parameters to be monitored or inspected except for steam generator divider plate assemblies, tube-to-tubesheet welds, heads (channel or lower/upper heads), and tubesheets. For these latter components, visual inspections are performed at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. These inspections of the steam generator head interior surfaces including the divider plate are intended to identify signs that cracking or loss of material may be occurring (e.g., through identification of rust stains).

4. ***Detection of Aging Effects:*** The ~~technical specifications~~ TSs require that a Steam Generator program be established and implemented to ensure that steam generator tube integrity is maintained. This requirement ensures that components that could compromise tube integrity are properly evaluated or monitored (e.g., degradation of a secondary side component that could result in a loss of tube integrity is managed by this program). The inspection requirements in the TSs ~~technical specifications~~ are intended to detect degradation (i.e., aging effects), if they should occur.

   The TSs ~~technical specifications~~ are performance-based, and the actual scope of the inspection and the expansion of sample inspections are justified based on the results of the inspections. The goal is to perform inspections at a frequency sufficient to provide reasonable assurance of steam generator tube integrity for the period of time between inspections.

The general condition of some components (e.g., plugs, ~~and~~ secondary side components, divider plates, and primary side cladding of channel heads and tubesheets) is monitored. It may be monitored visually, and, subsequently, more detailed inspections may be performed if degradation or evidence of degradation (e.g., rust stains) is detected.

NEI 97-06 provides additional guidance on inspection programs to detect degradation of tubes, sleeves, plugs, and secondary side internals. The frequencies of the inspections are based on technical assessments. Guidance on performing these technical assessments is contained in NEI 97-06 and the associated industry guidelines.

The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The EPRI PWR Steam Generator Examination Guidelines (EPRI ~~1013706~~ 3002007572) contains guidance on the qualification of steam generator tube inspection techniques.

The primary-to-secondary leakage monitoring program provides a potential indicator of a loss of steam generator tube integrity. NEI 97-06 and the associated EPRI guidelines provide information pertaining to an effective leakage monitoring program.

5. ***Monitoring and Trending:*** Condition monitoring assessments are performed to determine whether the structural- and accident-induced leakage performance criteria were satisfied during the prior operating interval. Operational assessments are performed to verify that structural and leakage integrity will be maintained for the planned operating interval before the next inspection. If tube integrity cannot be maintained for the planned operating interval before the next inspection, corrective actions are taken in accordance with the plant's corrective action program. Comparisons of the results of the condition monitoring assessment to the predictions of the previous operational assessment are performed to evaluate the adequacy of the previous operational assessment methodology. If the operational assessment was not conservative in terms of the number and/or severity of the condition, corrective actions are taken in accordance with the plant's corrective action program.

The ~~technical specifications~~ TSs require condition monitoring and operational assessments to be performed (although the ~~technical specifications~~ TSs do not explicitly require operational assessments, these assessments are necessary to ensure that the tube integrity will be maintained until the next inspection). Condition monitoring and operational assessments are done in accordance with the ~~technical specification~~ TS requirements and guidance in NEI 97-06 and the EPRI Steam Generator Integrity Assessment Guidelines.

The goal of the inspection program for all components covered by this AMP is to ensure that the components continue to function consistent with the design and licensing basis of the facility (including regulatory safety margins).

Assessments of the degradation ~~of steam generator secondary side internals~~ that may occur in the components covered by this AMP, except for steam generator divider plate assemblies, tube-to-tubesheet welds, heads, and tubesheets as noted above, are performed in accordance with the guidance in the EPRI Steam Generator Integrity Assessment Guidelines. All assessments of component degradation are performed to ensure the component continues to function consistent with the design and licensing basis and to ensure technical specification requirements are satisfied.

6. ***Acceptance Criteria:*** Assessment of tube and sleeve integrity and plugging or repair criteria of flawed and sleeved tubes is in accordance with plant ~~technical specifications~~ TSs. The criteria for plugging or repairing steam generator tubes and sleeves are based on U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.121 and are incorporated into

plant TSs ~~technical specifications~~. Guidance on assessing the acceptability of flaws is also provided in NEI 97-06 and the associated EPRI guidelines, including the EPRI PWR Steam Generator Examination Guidelines (EPRI 3002007572), EPRI Steam Generator In-Situ Pressure Test Guidelines (EPRI 1025132) and EPRI Steam Generator Integrity Assessment Guidelines (EPRI 3002007571).

Degraded plugs, divider plate assemblies, tube-to-tubesheet welds, heads (interior surfaces), tubesheets (primary side), ~~degraded~~ and secondary side internals~~, and leaving a loose part or a foreign object in the steam generator~~ are evaluated for continued acceptability on a case-by-case basis, as is leaving a loose part or a foreign object in a steam generator. NEI 97-06 and the associated EPRI guidelines provide guidance on the performance of some of these evaluations. The intent of all ~~these~~ evaluations is to ensure that the components will continue to perform their functions, ~~affected by parts or objects have adequate integrity~~ consistent with the design and licensing basis of the facility, and will not affect the integrity of other components (e.g., by generating loose parts).

Guidance on the acceptability of primary-to-secondary leakage and water chemistry parameters also are discussed in NEI 97-06 and the associated EPRI guidelines

7. ***Corrective Actions:*** Results that do not meet the acceptance criteria are addressed in the applicant's corrective action program under those specific portions of the quality assurance (QA) program that are used to meet Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B. The Appendix of the GALL Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the corrective actions element of this AMP for both safety-related and nonsafety-related structures and components (SCs) within the scope of this program.

   For degradation of steam generator tubes and sleeves (if applicable), the ~~technical specifications~~ TSs provide requirements on the actions to be taken when the acceptance criteria are not met. For degradation of other components, the appropriate corrective action is evaluated per NEI 97-06 and the associated EPRI guidelines, the American Society of Mechanical Engineers (ASME) Code Section XI (2004 Edition),[13] 10 CFR 50.65, and 10 CFR Part 50, Appendix B, as appropriate. ~~As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable for ensuring effective corrective actions.~~

8. ***Confirmation Process:*** The confirmation process is addressed through those specific portions of the QA program that are used to meet Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B. The Appendix of the GALL Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the confirmation process element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program. ~~Site quality assurance (QA) procedures, review and approval processes, and site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.~~

   ~~In addition, t~~ The adequacy of the preventive measures in the Steam Generator program is confirmed through periodic inspections.

9. ***Administrative Controls:*** ~~As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.~~ Administrative controls are addressed through the QA program that is used to meet

[13] Refer to the GALL Report, Chapter 1, for applicability of other editions of the ASME Code.

the requirements of 10 CFR Part 50, Appendix B, associated with managing the effects of aging. The Appendix of the GALL Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the administrative controls element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program.

10. ***Operating Experience:*** Several generic communications have been issued by the NRC related to the steam generator programs implemented at plants. The reference section lists many of these generic communications. In addition, NEI 97-06 provides guidance to the industry for routinely sharing pertinent steam generator operating experience and for incorporating lessons learned from plant operation into guidelines referenced in NEI 97-06. The latter includes providing interim guidance to the industry, when needed.

    The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals) such that the steam generators can perform their intended safety function.

    The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience.

**References**


10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, ~~2009~~ 2016.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, ~~2009~~ 2016.

10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, ~~2009~~ 2016.

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## Changes to GALL Report Tables IV.D1 and IV.D2

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

| Item | Link | Structure and/or Component | Material | Environment | Aging Effect/ Mechanism | Aging Management Program (AMP) | Further Evaluation |
|---|---|---|---|---|---|---|---|
| IV.D1.RP-367 | IV.D1-6(RP-21) | Primary side components: divider plate | Steel (with nickel-alloy cladding); nickel alloy | Reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" A plant-specific program is to be evaluated for nickel alloy divider plate assemblies and associated welds made of Alloy 600; the effectiveness of the ~~chemistry control~~ existing aging management programs should be verified to ensure that cracking due to PWSCC is not occurring if the conditions at the unit are not bounded by the industry analyses. | Yes, ~~detection of aging effects is to be evaluated~~ plant-specific. |
| IV.D1.RP-385 | | Tube-to-tube sheet welds | Nickel alloy | Reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" A plant-specific program is to be evaluated; the effectiveness of the ~~water chemistry~~ existing aging management programs should be verified to ensure cracking is not occurring if the conditions at the unit are not bounded by the industry analyses and approval has not been granted permanently for relocating the reactor coolant pressure boundary function from the welds. | Yes, plant-specific. |

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|---|---|---|---|---|---|---|---|
| IV.D1.R-436a | | Steam generator channel heads and tubesheets | Steel (with stainless steel or nickel alloy cladding) | Reactor coolant | Loss of material due to boric acid corrosion | AMP XI.M2, “Water Chemistry,” and AMP XI.M19, “Steam Generators” | No |

| IV | REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM | | | | | | |
|---|---|---|---|---|---|---|---|
| D2 | Steam Generator (Once-Through) | | | | | | |
| **Item** | **Link** | **Structure and/or Component** | **Material** | **Environment** | **Aging Effect/ Mechanism** | **Aging Management Program (AMP)** | **Further Evaluation** |
| IV.D2.RP-185 | IV.D2-4(R-35) | Tube-to-tube sheet welds | Nickel alloy | Reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, “Water Chemistry,” and Chapter XI.M19, “Steam Generators” A plant-specific program is to be evaluated; the effectiveness of the ~~water chemistry~~ existing aging management programs should be verified to ensure cracking is not occurring if the conditions at the unit are not bounded by the industry analyses and approval has not been granted permanently for relocating the reactor coolant pressure boundary function from the welds. | Yes, plant-specific |
| IV.D2.R-440a | | Steam generator upper and lower heads and tubesheets | Steel (with stainless steel or nickel alloy cladding) | Reactor coolant | Loss of material due to boric acid corrosion | AMP XI.M2, “Water Chemistry,” and AMP XI.M19, “Steam Generators” | No |

## Appendix C: Resolution of Public Comments

The Nuclear Energy Institute (NEI) submitted consolidated industry comments related to LR-ISG-2016-01 by letter dated July 7, 2016 (ADAMS Accession No. ML16194A026). No other public comment was submitted.

| No. | Section and Comment | Staff Resolution |
|---|---|---|
| 1 | Evaluation of PWSCC in Divider Plate Assemblies, second paragraph, first sentence, page 2:<br>Revise the sentence to read, "PWSCC can initiate under specific conditions ***and chromium content*** in high-strength nickel alloys ………" | The recommendation is incorporated. |
| 2 | Evaluation of PWSCC in Divider Plate Assemblies, last sentence of the 4th paragraph, page 2:<br>This area implies that the cracks occur in the stub runner. Yet the next paragraph last sentence says, "This new OE indicates that the cracks due to PWSCC in the divider plates remain shallow ….."<br>Suggest: "This new OE indicates that the cracks due to PWSCC in the divider plate ***assemblies*** remain shallow ….." | The recommendation is incorporated. |
| 3 | Evaluation of PWSCC in Divider Plate, last paragraph, page 3:<br>Implies there is a TLAA. This should be clarified. | The staff clarified that the analyses are not a TLAA. |
| 4 | Top of page 4:<br>The continuation of the paragraph is from page 3. The last two sentences change the topic from cracks in the divider plate assembly to cracks in the tube-to-tube sheet welds.<br>Recommend starting a new paragraph. | The recommendation is incorporated; however, please note that the topic of the paragraph is growth of flaws from the divider plate assembly into other components. The original paragraph addressed two components within the context of flaw growth analyses: steam generator shell and the tube-to-tubesheet welds. |
| 5 | 3rd full paragraph on page 4:<br>Rewrite paragraph to read, "Given the low likelihood of occurrence of cracking in the divider plate assembly and the lack of cracking progression, if it were to occur, analyses indicate that there are no structural integrity concerns. In addition there is no adverse effects on other analyses (e.g., tube repair criteria, tube repair methods, design basis analyses) if cracking were to occur. The staff concludes the following related to the on-going channel head visual inspections:" | The staff partially agrees with the comment. The recommendation is incorporated with a few editorial changes.<br>The discussed sentences are revised as follows:<br>In summary, operating experience indicates there is a low likelihood of occurrence of cracking in the divider plate assembly, and when cracking occurs, there is a general lack of progression. Industry analyses indicate that there are no structural integrity concerns associated with the cracking. In addition, there are no adverse effects on other analyses (e.g., tube repair criteria, tube repair methods, design basis analyses) if cracking were to occur. Therefore, the staff concludes the following related to aging management for divider plate assemblies: |

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| 6 | Evaluation of PWSCC in Divider Plate Assemblies, page 4, second bullet:<br><br>For units with divider plate assemblies fabricated with Alloy 600 or Alloy 600 weld materials, if the analyses performed by the industry (Reference 4) are applicable and bounding for the unit, the primary water chemistry program should be supplemented with a general visual inspection of the steam generator channel head (as part of the steam generator program as discussed in this LR-ISG). The purpose of the visual inspection is to identify rust stains or other abnormal conditions which could indicate the presence of cracking (e.g., distortion of divider plates). The visual inspection should be performed every time the channel head is accessed for steam generator tube inspections.<br><br>Comment: Based on the reports cited, the frequency should not be specified and left to the program.<br><br>Suggested words are as follows:<br><br>The applicant will perform general visual inspections of the divider plate assemblies looking for evidence of cracking (e.g., rust stains) as part of the steam generator program at least once per the period identified in Site Technical Specifications. | The staff partially agrees with the comment.<br><br>Since the inspection periods may vary based on tube materials, their heat treatment or other factors, the staff prescribes a consistent inspection interval for all units. Namely, each steam generator head is inspected at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. |
| 7 | Last paragraph on page 5:<br><br>It seems like there should be a section up front in the ISG that discusses all of the susceptibilities of nickel alloy 600 and 690, including this background and that from the next page. | The staff disagrees with the comment.<br><br>In staff's view, the structure of sections and arrangement of contents in the draft LR-ISG are sufficient to effectively convey proposed changes to the aging management guidance and their technical bases. Therefore, no change is made. |
| 8 | Last paragraph on page 7, second sentence:<br><br>States that primary to secondary leakage values assumed in the accident analysis could be managed.<br><br>Recommend including guidance for how it is managed (i.e., via primary leak rate determinations). | The staff partially agrees with the comment.<br><br>The recommendation is incorporated in the referenced sentence as follows:<br><br>…and that primary-to-secondary leakage through this joint could be limited to values assumed in the accident analyses (e.g., through testing, analyses, or monitoring operational leakage). |

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| 9 | Evaluation of Steam Generator Head and Tubesheet Degradation, page 9, third paragraph:<br><br>As a result, this LR-ISG revises GALL Report AMP XI.M19 to include steam generator primary side internal surfaces and to indicate that visual inspections of these surfaces should be performed each time the steam generator is accessed for tube inspections.<br><br><u>Comment</u>: Based on the reports cited, the frequency should not be specified and left to the program.<br><br><u>Suggested words are as follows</u>:<br><br>The applicant will perform general visual inspections of the primary side internal surfaces looking for evidence of cracking (e.g., rust stains) as part of the steam generator program at least once per the period identified in site Technical Specifications related to the on-going channel head visual inspections.<br><br><u>Commenter's Note</u>:<br><br>Example: Both plants have alloy 690 tubes. One plant inspects the tubes every 2 years. The other plant can skip and inspects the tubes every 4.2 years. Therefore one plant inspects the cited component twice as often as the other plant. | The staff partially agrees with the comment. The recommendation is incorporated with additional changes.<br><br>The resolution of Comment 6 addresses the same topic. The last sentence of the third paragraph is revised as follows:<br><br>As a result, this LR-ISG revises GALL Report AMP XI.M19 to include steam generator primary side internal surfaces and to indicate that visual inspections of these surfaces should be performed at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. |
| 10 | Section 3.1.2.2.11.1, page 13:<br><br>Last sentence of the first paragraph. Remove "of the divider plate thickness." The crack is in the sub runner not the divider plate. Also in the last line on this page add "assembly" after "divider plate." | The recommendation is incorporated. |
| 11 | Table 3.0-1, page 19:<br><br>Revise introductory text at top of the page to correct a typo as follows:<br><br>"As addressed in LR-ISG-2011-02….GALL Report AMP ~~XI.M16~~XI.M19…" | The recommendation is incorporated since XI.M19 is the correct reference. |

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| 12 | Appendix A, Table 3.0-1, page 19:<br><br>The table states in part "This program also performs visual inspections of steam generator heads (internal areas) when they are accessed for tube inspections, to manage the aging of divider plate assemblies (as applicable), tube to tubesheet welds, heads (interior surfaces) and tubesheets (primary side).<br><br>Comment: Based on the reports cited, the frequency should not be specified and left to the program.<br><br>Suggested words are as follows:<br><br>The applicant will perform general visual inspections of the steam generator heads looking for evidence of cracking (e.g., rust stains) as part of the steam generator program at least once per the period identified in Site Technical Specifications. | The staff partially agrees with the comment.<br><br>The staff finds that the visual inspections look for evidence of loss of material as well as evidence for cracking. The recommendation is incorporated with additional changes. The referenced sentence is revised as follows:<br><br>This program also performs general visual inspections of the steam generator heads (internal surfaces) looking for evidence of cracking or loss of material (e.g., rust stains) at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. |
| 13 | Changes to SRP-LR Table 3.0-1 (FSAR supplement), page 19:<br><br>As addressed in LR-ISG-2011-02, "Aging Management Program for Steam Generators," GALL Report AMP XI.M16 relies on the guidance in Revision 3 of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and EPRI Steam Generator Integrity Assessment Guidelines (EPRI 1019038).<br><br>Comment: The Report number referenced refers to Rev. 3. Rev. 4 is due for issue in June 2016, (EPRI Report # 3002007571).<br><br>Suggested words are as follows:<br><br>As addressed in LR-ISG-2011-02, "Aging Management Program for Steam Generators," GALL Report AMP XI.M16 relies on the guidance in Revision 3 of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and EPRI Steam Generator Integrity Assessment Guidelines~~ (EPRI 1019038)~~.<br><br>Commenter's Note:<br><br>EPRI SG Guidelines are revised every 5-8 years and typically require mandatory implementation by the PWR utility within 1 year of issue. By LR [license renewal] locking-in obsolete guideline revisions an immediate discrepancy between LR commitments and SG Guidelines occurs. All PWR Sites have committed through NEI 97-06 to work to SGMP EPRI Guidelines (current revisions) | The staff partially agrees with the comment.<br><br>The staff agrees that there is a need to consider Revision 4 of the EPRI Steam Generator Integrity Assessment Guidelines in developing the FSAR supplement summary description.<br><br>In contrast, the staff disagrees with the comment that specific revisions of EPRI guidelines should be deleted from the aging management program and this LR-ISG. The staff notes that identification of the specific revisions of guidelines incorporated into the program is important and necessary to clearly define program elements and attributes.<br><br>The staff also notes that well-defined program elements establish solid bases upon which further program enhancements can be made as necessary in adequate consideration of more recent guidelines and operating experience.<br><br>Therefore, the introductory sentences for the FSAR supplement table are revised to reflect the comment resolution as follows:<br><br>As addressed in LR-ISG-2011-02, "Aging Management Program for Steam Generators," GALL Report AMP XI.M19 relies on the guidance in Revision 3 of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines." This change with updated references to industry steam generator guidelines is reflected in the following table on the FSAR supplement for GALL AMP XI.M19 and in Appendix B of this LR-ISG. |

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| 14 | Appendix B: Changes to the Guidance in NUREG-1801 (GALL Report), Revision 2, page 20:<br><br>In this appendix, the strikeout indicates where deletions are made and the underscore indicates where additions are made. The changes to GALL Report AMP XI.M19 and AMR tables are provided below. As discussed in Appendix A of this LR-ISG, the changes addressed in LR-ISG-2011-02 (i.e., updated reference to Revision 3 of NEI 97-06 and correction in referencing the EPRI integrity assessment guidelines, EPRI 1019038) are also reflected in this appendix.<br><br><u>Comment</u>: the Report number referenced refers to Rev. 3. Rev. 4 is due for issue in June 2016, (EPRI Report # 3002007571).<br><br><u>Suggested words are as follows</u>:<br><br>In this appendix, the strikeout indicates where deletions are made and the underscore indicates where additions are made. The changes to GALL Report AMP XI.M19 and AMR tables are provided below. As discussed in Appendix A of this LR-ISG, the changes addressed in LR-ISG-2011-02 (i.e., updated reference to Revision 3 of NEI 97-06 and correction in referencing the EPRI integrity assessment guidelines~~, EPRI 1019038)~~ are also reflected in this appendix. | As previously discussed, the staff disagrees with the comment recommending that specific revisions of EPRI guidelines be deleted from the program and this LR-ISG. Please see the resolution of Comment 13 that addresses the same topic. The introductory sentences for the changes to the guidance in the GALL Report are revised to reflect the comment resolution as follows:<br><br>As discussed in Appendix A of this LR-ISG, the change addressed in LR-ISG-2011-02 (i.e., updated reference to Revision 3 of NEI 97-06) and updated references to the EPRI steam generator guidelines are also reflected in this appendix. |
| 15 | AMP XI.M19 (Introduction), page 20, also References, page 25:<br><br>Revision 3 of NEI 97-06 is a guideline document that is updated periodically. Reference to a specific revision results in an inconsistency that could be eliminated by referring to the “latest issued revision” or not calling out a revision at all. | The staff disagrees with the comment as previously discussed. The staff notes that identification of specific revisions of industry guidance incorporated into the program is important and necessary to clearly define program elements and attributes.<br><br>The staff also notes that well-defined program elements establish solid bases upon which further program enhancements can be made as necessary in adequate consideration of more recent guidelines and operating experience<br><br>Please see the resolution of Comment 13 that addresses the same topic. |

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Program Element 3, page 22:

It is recommended that GALL text eliminate reference to specific EPRI Report numbers. Instead, consider one of the following text change options:

1. "The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines ~~(EPRI 1008219)~~ <u>are periodically updated based on industry operating experience and</u> provide~~s~~ guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines ~~(EPRI 1012987 1019038)~~ <u>are periodically updated based on industry operating experience and</u> provide guidance on secondary side activities."

OR

2. "The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines ~~(EPRI 1008219) provides guidance on monitoring primary-to-secondary leakage. The~~ <u>and</u> EPRI Steam Generator Integrity Assessment Guidelines ~~(EPRI 1012987 1019038)~~ <u>are periodically updated based on industry operating experience and</u> provide guidance on <u>monitoring primary to secondary leakage and</u> secondary side activities<u>, respectively</u>."

<u>Commenter's Note</u>:

Statements on operating experience being recommended, are intended to align with the new text added to Element 10:

"The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience."

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As previously discussed, the staff disagrees with the comment recommending that specific revisions of EPRI guidelines be deleted from the program. Please see the resolution of Comment 13 that addresses the same topic

As discussed in the resolutions of Comments 13 and 17, the program, including the references section, is revised to incorporate the following updated references:

- Revision 4 of EPRI Steam Generator Integrity Assessment Guidelines, EPRI 3002007571, June 2016
- Revision 4 of EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines, EPRI 1022832, November 2011

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| 17 | Program Element 3, 5th paragraph, page 22:<br><br>Water chemistry parameters are also monitored and controlled, as discussed in AMP XI.M2. The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines (EPRI 1008219) provides guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines (EPRI ~~1012987~~ 1019038) provide guidance on secondary side activities.<br><br><u>Comments</u>: EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines (EPRI 1008219). Report number referenced refers to Rev. 3. Rev. 4 was issued Sept, 2011, (EPRI Report #1022832). EPRI Steam Generator Integrity Assessment Guidelines (EPRI ~~1012987~~ 1019038) provide guidance on secondary side activities. Report number referenced refers to Rev. 3. Rev. 4 will be issued June, (EPRI Report # 3002007571).<br><br><u>Suggested words are as follows</u>:<br><br>Water chemistry parameters are also monitored and controlled, as discussed in AMP XI.M2. The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines ~~(EPRI 1008219)~~ provides guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines ~~(EPRI 1012987 1019038)~~ provide guidance on secondary side activities.<br><br><u>Commenter's Note</u>:<br><br>EPRI SG Guidelines are revised every 5-8 years and typically require mandatory implementation by the PWR utility within 1 year of issue. By LR [license renewal] locking-in obsolete guideline revisions an immediate discrepancy between LR commitments and SG Guidelines occurs. All PWR Sites have committed through NEI 97-06 to work to SGMP EPRI Guidelines (current revisions). | The staff agrees with the comment that the most recent revision of EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines is Revision 4. One note is that the most recent revision of the guidelines was published in November 2011. As previously discussed, the staff disagrees with the comment recommending that specific revisions of EPRI guidelines be deleted from the program. Please see the resolution of Comment 13 that addresses the same topic.<br><br>Based on this discussion, the references to the guidelines in the program (including the references section) are updated accordingly. |
| 18 | Program Element 4, 5th paragraph, page 23:<br><br>The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The EPRI PWR Steam Generator Examination Guidelines (EPRI 1013706) contains guidance on the qualification of steam generator tube inspection techniques.<br><br><u>Comment</u>: EPRI PWR Steam Generator Examination Guidelines (EPRI 1013706). Report number referenced refers to Rev. 7. Rev. 8 will be issued June, 2016, (EPRI Report # 3002007572).<br><br><u>Suggested words are as follows</u>:<br><br>The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The EPRI PWR Steam Generator Examination Guidelines ~~(EPRI 1013706)~~ contains guidance on the qualification of steam generator tube inspection techniques. | The staff partially agrees with the comment. As previously discussed, the staff agrees with the comment that the most recent revisions of the EPRI guidelines need to be considered in developing this LR-ISG.<br><br>The staff disagrees with the comment recommending that specific revisions of EPRI guidelines be deleted from the program. Please see the resolution of Comment 13 that addresses the same topic.<br><br>The commented sentence (along with the references section) is revised to refer to Revision 8 of the examination guidelines (EPRI 3002007572) as follows:<br><br>The EPRI PWR Steam Generator Examination Guidelines (EPRI 3002007572) contains guidance on the qualification of steam generator tube inspection techniques. |

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| 19 | References, page 25:<br><br>Add references to EPRI 1013706 (currently mentioned in Element 4) and to EPRI 1025132, EPRI Steam Generator In-Situ Pressure Testing Guidelines, Revision 4, as mentioned at the top of page 24 under Element 6. | The staff partially agrees with the comment.<br><br>As discussed in the resolution of Comment 18, the most recent version of the EPRI Steam Generator Examination Guidelines is Revision 8 (EPRI 3002007572) rather than Revision 7 (EPRI 1013706). The staff agrees with the comment regarding Revision 4 of EPRI Steam Generator In-Situ Pressure Testing Guidelines.<br><br>The references section is revised to reflect these updates. |
| 20 | References, 4th Reference, page 25:<br><br>EPRI 1008219, PWR Primary-to-Secondary Leak Guidelines: Revision 3, Electric Power Research Institute, Palo Alto, CA, December 2004.<br><br>Comment: Report number referenced refers to Rev. 3. Rev. 4 was issued Sept, 2011 (EPRI Report #1022832).<br><br>Suggested words are as follows:<br><br>EPRI ~~1008219~~, PWR Primary-to-Secondary Leak Guidelines~~: Revision 3~~, Electric Power Research Institute, Palo Alto, CA, ~~December 2004~~(current revision). | The staff partially agrees with the comment. The staff agrees with the comment recommending that Revision 4 of EPRI PWR Primary-to-Secondary Leak Guidelines needs to be considered in developing this LR-ISG. However, the staff disagrees with the recommendation that specific revisions of EPRI guidelines be deleted from the program. Please see the resolutions of Comments 13 and 17 that address the same topic.<br><br>Based on this discussion, the references section is revised to refer to Revision 4 of EPRI PWR Primary-to-Secondary Leak Guidelines (EPRI 1022832). |
| 21 | References, 5th Reference, page 25:<br><br>EPRI ~~1012987~~ 1019038, Steam Generator Integrity Assessment Guidelines: Revision ~~2~~ 3, Electric Power Research Institute, Palo Alto, CA, ~~July 2006~~ November 2009.<br><br>Comment: Report number referenced refers to Rev. 3. Rev. 4 will be issued June, 2016, (EPRI Report #3002007571).<br><br>Suggested words are as follows:<br><br>EPRI ~~1012987 1019038~~, Steam Generator Integrity Assessment Guidelines: ~~Revision 2 3~~, Electric Power Research Institute, Palo Alto, CA, ~~July 2006 November 2009~~ (current revision).<br><br>Commenter's Note:<br><br>EPRI SG Guidelines are revised every 5-8 years and typically require mandatory implementation by the PWR utility within 1 year of issue. By LR locking-in obsolete guideline revisions an immediate discrepancy between LR commitments and SG Guidelines occurs. All PWR Sites have committed through NEI 97-06 to work to SGMP EPRI Guidelines (current revisions). | The staff partially agrees with the comment. The staff agrees with the comment that Revision 4 of EPRI Steam Generator Integrity Assessment Guidelines needs to be considered in developing this LR-ISG. However, the staff disagrees with the comment recommending that specific revisions of EPRI guidelines be deleted from the program. Please see the resolution of Comment 13 that addresses the same topic.<br><br>Based on this discussion, the references section is revised to refer to Revision 4 of EPRI Steam Generator Integrity Assessment Guidelines (EPRI 3002007571). |

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| 22 | Changes to GALL Table IV.D1, page 27:<br><br>For item IV.D1.RP-367, since a plant specific program is specified, the description of the AMP should be, "<u>A plant-specific program is to be evaluated</u> ~~F~~*f*or nickel alloy divider plate assemblies and associated welds made of Alloy 600; ~~if the conditions at the unit are not bounded by the industry analyses,~~ the effectiveness of the existing aging management programs should be verified to ensure that cracking due to PWSCC is not occurring <u>if the conditions at the unit are not bounded by the industry analyses</u>." | The staff agrees with the comment. The recommendation is incorporated. |

**Appendix D: Revised Guidance in NUREG-1800 (SRP-LR), Revision 2**

**Revised Section 3.1.2.2.11.1 (Acceptance Criteria for Divider Plate Assemblies)**

Foreign operating experience in steam generators with a design similar to that of Westinghouse steam generators (particularly Model 51) has identified cracks due to primary water stress corrosion cracking (PWSCC) in steam generator (SG) divider plate assemblies fabricated of Alloy 600 and/or the associated Alloy 600 weld materials, even with proper primary water chemistry. Cracks have been detected in the stub runner with depths typically about 0.08 inches (EPRI 3002002850).

All but one of these instances of cracking have been detected in divider plate assemblies that are approximately 1.3 inches in thickness. For the cracks in the 1.3-inch thick divider plate assemblies, the cracks tend to be parallel to the divider-plate-to-stub-runner weld (i.e., run horizontally in parallel to the lower surface of the tubesheet). For the one instance of cracking in a divider plate assembly with a thickness greater than 1.3 inches, the cracking occurred in a divider plate assembly with a thickness of approximately 2.4 inches near manufacturing marks on the upper end of the stub runner used for locating tubesheet holes. These flaws were estimated to be approximately 0.08-inch deep.

Although these instances indicate that the water chemistry program may not be sufficient to manage cracking due to PWSCC in SG divider plate assemblies, analyses by the industry indicate that PWSCC in the divider plate assembly does not pose a structural integrity concern for other steam generator components (e.g., tubesheet and tube-to-tubesheet welds) and does not adversely affect other safety analyses (e.g., analyses supporting tube plugging and repairs, tube repair criteria, and design basis accidents). In addition, the industry analyses indicate that flaws in the divider plate assembly will not adversely affect the heat transfer function (as a result of bypass flow) during normal forced flow operation, during natural circulation conditions (assessed in the analyses of various design basis accidents), or in the event of a loss-of-coolant accident (LOCA).

Furthermore, additional industry analyses indicate that PWSCC in the divider plate assembly is unlikely to adversely impact adjacent items, such as the tubesheet cladding, tube-to-tubesheet welds, and channel head. Therefore,

- For units with divider plate assemblies fabricated of Alloy 690 and Alloy 690 type weld materials, a plant-specific aging management program (AMP) is not necessary.

- For units with divider plate assemblies fabricated of Alloy 600 or Alloy 600 type weld materials, if the analyses performed by the industry (EPRI 3002002850) are applicable and bounding for the unit, a plant-specific AMP is not necessary.

- For units with divider plate assemblies fabricated of Alloy 600 or Alloy 600 type weld materials, if the industry analyses (EPRI 3002002850) are not bounding for the applicant's unit, a plant-specific AMP is necessary or a rationale is necessary for why such a program is not needed. A plant-specific AMP (one beyond the primary water chemistry and the steam generator programs) may include a one-time inspection that is capable of detecting cracking to verify the effectiveness of the water chemistry and steam generator programs and the absence of PWSCC in the divider plate assemblies.

The existing programs rely on control of reactor water chemistry to mitigate cracking due to PWSCC and general visual inspections of the channel head interior surfaces (included as part of the steam generator program). The GALL Report recommends further evaluation for a plant-specific AMP to confirm the effectiveness of the primary water chemistry and steam generator programs as described in this section. Acceptance criteria for a plant-specific AMP are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR). In place of a plant-specific AMP, the applicant may provide a rationale to justify why a plant-specific AMP is not necessary.

**Revised Section 3.1.3.2.11.1 (Review Procedures for Divider Plate Assemblies)**

The GALL Report recommends further evaluation of a plant-specific AMP, along with the primary water chemistry and steam generator programs, to manage cracking due to PWSCC in nickel alloy divider plate assemblies. For divider plate assemblies fabricated of Alloy 690 and Alloy 690 type welding materials, a plant-specific AMP is not necessary. For divider plate assemblies made of Alloy 600 or Alloy 600 type welding materials, the reviewer verifies that the applicant has an adequate basis for concluding that the analyses performed by the industry (EPRI 3002002850) assessing the significance of divider plate cracking are applicable and bounding for the conditions at its unit. If the industry's analyses are not bounding, the reviewer evaluates the applicant's plant-specific AMP on a case-by-case basis to ensure that an adequate plant-specific AMP will be in place for the management of this aging effect or the reviewer reviews the applicant's rationale (e.g., a more detailed plant-specific evaluation than performed by the industry) for why such a plant-specific AMP is not necessary.

**Revised Section 3.1.2.2.11.2 (Acceptance Criteria for Tube-to-Tubesheet Welds)**

Cracking due to PWSCC could occur in steam generator nickel alloy tube-to-tubesheet welds exposed to reactor coolant. The acceptance criteria for this review are:

- For units with Alloy 600 steam generator tubes and for which an alternate repair criteria such as C*, F*, W*, or H* has been permanently approved for both the hot- and cold-leg side of the steam generator, the weld is no longer part of the reactor coolant pressure boundary and a plant-specific AMP is not necessary;

- For units with Alloy 600 steam generator tubes, if there is no permanently approved alternate repair criteria such as C*, F*, W*, or H* or permanent approval applies to only either the hot- or cold-leg side of the steam generator, a plant-specific AMP is necessary;

- For units with thermally treated Alloy 690 steam generator tubes and with tubesheet cladding using Alloy 690 type material, a plant-specific AMP is not necessary;

- For units with thermally treated Alloy 690 steam generator tubes and with tubesheet cladding using Alloy 600 type material, a plant-specific AMP is necessary unless the applicant confirms that the industry's analyses for tube-to-tubesheet weld cracking (e.g., chromium content for the tube-to-tubesheet welds is approximately 22 percent and the tubesheet primary face is in compression as discussed in EPRI 3002002850) are applicable and bounding for the unit, and the applicant will perform general visual inspections of the tubesheet region looking for evidence of cracking (e.g., rust stains on the tubesheet cladding) as part of the steam generator program. In lieu of a plant-specific AMP, the applicant may provide a rationale for why a plant-specific AMP is not necessary.

The existing programs rely on control of reactor water chemistry to mitigate cracking due to PWSCC and visual inspections of the steam generator head interior surfaces. Along with the primary water chemistry and steam generator programs, the GALL Report recommends further evaluation of a plant-specific AMP to confirm the effectiveness of the primary water chemistry and steam generator programs in certain circumstances. A plant-specific AMP may include a one-time inspection that is capable of detecting cracking to confirm the absence of PWSCC in the tube-to-tubesheet welds. Acceptance criteria for a plant-specific AMP are described in Branch Technical Position RLSB-1 (Appendix A.1 of this SRP-LR). In place of a plant-specific AMP, the applicant may provide a rationale to justify why a plant-specific AMP is not necessary.

**Revised Section 3.1.3.2.11.2 (Review Procedures for Tube-to-Tubesheet Welds)**

The GALL Report recommends further evaluation for a plant-specific AMP, along with the primary water chemistry and steam generator programs, to manage cracking due to PWSCC in nickel alloy steam generator tube-to-tubesheet welds exposed to reactor coolant. The reviewer verifies the combination of materials of construction of the steam generator tubes and tubesheet cladding and the classification of the tube-to-tubesheet weld (i.e., whether it is part of the reactor coolant pressure boundary). If the combination and classification require a plant-specific AMP, the reviewer reviews the applicant's proposed program on a case-by-case basis to ensure adequate management of this aging effect. Alternatively, the reviewer evaluates applicant's rationale for why a plant-specific AMP is not necessary.

**Changes to Section 3.1.6**

Section 3.1.6 is revised to add the following references:


31. EPRI 3002002850, "Steam Generator Management Program: Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly," Electric Power Research Institute, Palo Alto, CA, October 2014.

32. EPRI 1020988, "Steam Generator Management Program: Phase II Divider Plate Cracking Engineering Study," Electric Power Research Institute, Palo Alto, CA, November 2010.

33. EPRI 1016552, "Divider Plate Cracking in Steam Generators: Results of Phase II: Evaluation of the Impact of a Cracked Divider Plate on LOCA and Non-LOCA Analyses," Electric Power Research Institute, Palo Alto, CA, November 2008.

34. EPRI Presentation, "NRC/Industry Meeting Regarding Tube-to-Tubesheet Weld and Divider Plate Cracking Report," July 30, 2015.

**Revised SRP-LR Table 3.1-1 (AMR Items)**

| Table 3.1-1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report | | | | | | | |
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| **ID** | **Type** | **Component** | **Aging Effect/Mechanism** | **Aging Management Programs** | **Further Evaluation Recommended** | **Rev2 Item** | **Rev1 Item** |
| 25 | PWR | Steel (with nickel-alloy cladding) or nickel alloy steam generator primary side components: divider plate and tube-to-tube sheet welds exposed to reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and<br>Chapter XI.M19, "Steam Generators" | Yes, plant-specific (See subsections 3.1.2.2.11.1 and 3.1.2.2.11.2) | IV.D1.RP-367<br>IV.D1.RP-385<br>IV.D2.RP-185 | IV.D1-6(RP-21)<br>N/A<br>IV.D2-4(R-35) |
| 127a | PWR | Steel (with stainless steel or nickel alloy cladding) steam generator heads and tubesheets exposed to reactor coolant | Loss of material due to boric acid corrosion | Chapter XI.M2, "Water Chemistry," and<br>Chapter XI.M19, "Steam Generators" | No | IV.D1.R-436a<br>IV.D2.R-440a | N/A |

**Revised SRP-LR Table 3.0-1, FSAR supplement for GALL Report AMP XI.M19**

| Table 3.0-1 FSAR Supplement for Aging Management of Applicable Systems | | | | |
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| **GALL Chapter** | **GALL Program** | **Description of Program** | **Implementation Schedule*** | **Applicable GALL Report and SRP-LR Chapter References** |
| XI.M19 | Steam Generators | This program manages the aging of steam generator tubes, plugs, sleeves, divider plate assemblies (as applicable), tube-to-tubesheet welds, heads (interior surfaces of channel or lower/upper heads), tubesheets (primary side), and secondary side components that are contained within the steam generator. This program consists of aging management activities for the steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator in accordance with the plant technical specifications and includes commitments to NEI 97-06, Revision 3 and the associated EPRI guidelines. This program also performs general visual inspections of the steam generator heads (internal surfaces) looking for evidence of cracking or loss of material (e.g., rust stains) at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. The program includes foreign material exclusion as a means to inhibit wear degradation, and secondary side maintenance activities, such as sludge lancing, for removing deposits that may contribute to component degradation. The program performs volumetric examination on steam generator tubes in accordance with the requirements in the technical specifications to detect aging effects, if they should occur. The technical specifications require condition monitoring (explicitly) and operational assessments (implicitly) to be performed to ensure that the tube integrity will be maintained until the next inspection. | Existing program | GALL IV / SRP 3.1 |

## XI.M19 STEAM GENERATORS

### Program Description

The Steam Generator program is applicable to managing the aging of steam generator tubes, plugs, sleeves, divider plate assemblies, tube-to-tubesheet welds, heads (interior surfaces of channel or lower/upper heads), tubesheet(s) (primary side), and secondary side components that are contained within the steam generator (i.e., secondary side internals). The aging of steam generator pressure vessel welds is managed by other programs such as GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry."

The establishment of a steam generator program for ensuring steam generator tube integrity is required by plant technical specifications (TSs). The steam generator tube integrity portion of the TSs at each PWR contains the same fundamental requirements as outlined in the standard technical specifications of NUREG-1430, Volume 1, Rev. 4, for Babcock & Wilcox (B&W) PWRs; NUREG-1431, Volume 1, Rev. 4, for Westinghouse PWRs; and NUREG-1432, Volume 1, Rev. 4, for Combustion Engineering PWRs. The requirements pertaining to steam generators in these three versions of the standard TSs are essentially identical. The TSs require tube integrity to be maintained and specify performance criteria, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging or repair of flawed tubes, acceptable tube repair methods, and leakage monitoring requirements.

The nondestructive examination techniques used to inspect steam generator components covered by this program are intended to identify components (e.g., tubes, plugs) with degradation that may need to be removed from service (e.g., tubes), repaired, or replaced, as appropriate.

The Steam Generator program at PWRs is modeled after Nuclear Energy Institute (NEI) 97-06, Revision 3, "Steam Generator Program Guidelines." This program references a number of industry guidelines (e.g., the Electric Power Research Institute (EPRI) PWR Steam Generator Examination Guidelines, PWR Primary-to-Secondary Leak Guidelines, PWR Primary Water Chemistry Guidelines, PWR Secondary Water Chemistry Guidelines, Steam Generator Integrity Assessment Guidelines, Steam Generator In Situ Pressure Test Guidelines) and incorporates a balance of prevention, mitigation, inspection, evaluation, repair, and leakage monitoring measures. The NEI 97-06 document (a) includes performance criteria that are intended to provide assurance that tube integrity is being maintained consistent with the plant's licensing basis and (b) provides guidance for monitoring and maintaining the tubes to provide assurance that the performance criteria are met at all times between scheduled inspections of the tubes. Steam generator tube integrity can be affected by degradation of steam generator plugs, sleeves, and secondary side components. The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves and secondary side components.

Degradation of divider plate assemblies, tube-to-tubesheet welds, heads (internal surfaces), or tubesheets (primary side) may have safety implications. Therefore, all of these components and the steam generator tubes, plugs, sleeves and secondary side components are addressed by this aging management program (AMP).

**Evaluation and Technical Basis**

1. ***Scope of Program:*** This program addresses degradation associated with steam generator tubes, plugs, sleeves, divider plate assemblies, tube-to-tubesheet welds, heads (interior surfaces of channel or lower/upper heads), tubesheet(s) (primary side), and secondary side components that are contained within the steam generator (i.e., secondary side internals). The program does not cover the steam generator secondary side shell, any nozzles attached to the secondary side shell or steam generator head, or the welds associated with these components. In addition, the program does not cover steam generator head welds (other than general corrosion of these welds caused as a result of degradation (defects/flaws) in the primary side cladding).

2. ***Preventive Actions:*** This program includes preventive and mitigative actions for addressing degradation. Preventive and mitigative measures that are part of the Steam Generator program include foreign material exclusion programs, and other primary and secondary side maintenance activities. The program includes foreign material exclusion as a means to inhibit wear degradation and secondary side maintenance activities, such as sludge lancing, for removing deposits that may contribute to degradation. Guidance on foreign material exclusion is provided in NEI 97-06. Guidance on maintenance of secondary side integrity is provided in the EPRI Steam Generator Integrity Assessment Guidelines. Primary side preventive maintenance activities include replacing plugs made with corrosion susceptible materials with more corrosion resistant materials and preventively plugging tubes susceptible to degradation.

   Extensive deposit buildup in the steam generators could affect tube integrity. The EPRI Steam Generator Integrity Assessment Guidelines, which are referenced in NEI 97-06, provide guidance on maintenance on the secondary side of the steam generator, including secondary side cleaning. Secondary side water chemistry plays an important role in controlling the introduction of impurities into the steam generator and potentially limiting their deposition on the tubes. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Water chemistry is monitored and maintained in accordance with the Water Chemistry program. The program description and evaluation and technical basis of monitoring and maintaining water chemistry are addressed in GALL AMP XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** There are currently three types of steam generator tubing used in the United States: mill annealed Alloy 600, thermally treated Alloy 600, and thermally treated Alloy 690. Mill annealed Alloy 600 steam generator tubes have experienced degradation due to corrosion (e.g., primary water stress corrosion cracking, outside diameter stress corrosion cracking, intergranular attack, pitting, and wastage) and mechanically induced phenomena (e.g., denting, wear, impingement damage, and fatigue). Thermally treated Alloy 600 steam generator tubes have experienced degradation due to corrosion (primarily cracking) and mechanically induced phenomena (primarily wear). Thermally treated Alloy 690 tubes have only experienced tube degradation due to mechanically induced phenomena (primarily wear).

   Degradation of tube plugs, sleeves, heads, tubesheet(s), and secondary side internals has also been observed, depending, in part, on the material of construction of the specific component. The potential for degradation exists for divider plate assemblies and tube-to-tubesheet welds, depending, in part, on the materials of construction for the components.

   The program includes an assessment of the forms of degradation to which a component is susceptible and implementation of inspection techniques capable of detecting those forms of degradation. The parameter monitored is specific to the component and the acceptance criteria for the inspection. For example, the severity of tube degradation may be evaluated in terms of the depth of degradation or measured voltage, dependent on whether a depth-based or voltage-based tube repair criteria (acceptance criteria) is being implemented for that specific degradation mechanism. Other parameters monitored include signals of excessive deposit buildup (e.g., steam generator water level oscillations), which may result in fatigue failure of tubes or corrosion of the tubes; water chemistry parameters, which may indicate unacceptable levels of impurities; primary-to-secondary leakage, which may indicate excessive tube, plug, or sleeve degradation; and the presence of loose parts or foreign objects on the primary and secondary side of the steam generator, which may result in tube damage.

Water chemistry parameters are also monitored and controlled, as discussed in AMP XI.M2. The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines (EPRI 1022832) provides guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines (EPRI 3002007571) provide guidance on secondary side activities.

In summary, the NEI 97-06 program provides guidance on parameters to be monitored or inspected except for steam generator divider plate assemblies, tube-to-tubesheet welds, heads (channel or lower/upper heads), and tubesheets. For these latter components, visual inspections are performed at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections. These inspections of the steam generator head interior surfaces including the divider plate are intended to identify signs that cracking or loss of material may be occurring (e.g., through identification of rust stains).

4. ***Detection of Aging Effects:*** The TSs require that a Steam Generator program be established and implemented to ensure that steam generator tube integrity is maintained. This requirement ensures that components that could compromise tube integrity are properly evaluated or monitored (e.g., degradation of a secondary side component that could result in a loss of tube integrity is managed by this program). The inspection requirements in the TSs are intended to detect degradation (i.e., aging effects), if they should occur.

   The TSs are performance-based, and the actual scope of the inspection and the expansion of sample inspections are justified based on the results of the inspections. The goal is to perform inspections at a frequency sufficient to provide reasonable assurance of steam generator tube integrity for the period of time between inspections.

   The general condition of some components (e.g., plugs, secondary side components, divider plates, and primary side cladding of channel heads and tubesheets) is monitored. It may be monitored visually, and, subsequently, more detailed inspections may be performed if degradation or evidence of degradation (e.g., rust stains) is detected.

   NEI 97-06 provides additional guidance on inspection programs to detect degradation of tubes, sleeves, plugs, and secondary side internals. The frequencies of the inspections are based on technical assessments. Guidance on performing these technical assessments is contained in NEI 97-06 and the associated industry guidelines.

   The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The EPRI PWR Steam Generator Examination Guidelines (EPRI 3002007572) contains guidance on the qualification of steam generator tube inspection techniques.

The primary-to-secondary leakage monitoring program provides a potential indicator of a loss of steam generator tube integrity. NEI 97-06 and the associated EPRI guidelines provide information pertaining to an effective leakage monitoring program.

5. ***Monitoring and Trending:*** Condition monitoring assessments are performed to determine whether the structural- and accident-induced leakage performance criteria were satisfied during the prior operating interval. Operational assessments are performed to verify that structural and leakage integrity will be maintained for the planned operating interval before the next inspection. If tube integrity cannot be maintained for the planned operating interval before the next inspection, corrective actions are taken in accordance with the plant's corrective action program. Comparisons of the results of the condition monitoring assessment to the predictions of the previous operational assessment are performed to evaluate the adequacy of the previous operational assessment methodology. If the operational assessment was not conservative in terms of the number and/or severity of the condition, corrective actions are taken in accordance with the plant's corrective action program.

   The TSs require condition monitoring and operational assessments to be performed (although the TSs do not explicitly require operational assessments, these assessments are necessary to ensure that the tube integrity will be maintained until the next inspection). Condition monitoring and operational assessments are done in accordance with the TS requirements and guidance in NEI 97-06 and the EPRI Steam Generator Integrity Assessment Guidelines.

   The goal of the inspection program for all components covered by this AMP is to ensure that the components continue to function consistent with the design and licensing basis of the facility (including regulatory safety margins).

   Assessments of the degradation that may occur in the components covered by this AMP, except for steam generator divider plate assemblies, tube-to-tubesheet welds, heads, and tubesheets as noted above, are performed in accordance with the guidance in the EPRI Steam Generator Integrity Assessment Guidelines. All assessments of component degradation are performed to ensure the component continues to function consistent with the design and licensing basis and to ensure technical specification requirements are satisfied.

6. ***Acceptance Criteria:*** Assessment of tube and sleeve integrity and plugging or repair criteria of flawed and sleeved tubes is in accordance with plant TSs. The criteria for plugging or repairing steam generator tubes and sleeves are based on U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.121 and are incorporated into plant TSs. Guidance on assessing the acceptability of flaws is also provided in NEI 97-06 and the associated EPRI guidelines, including the EPRI PWR Steam Generator Examination Guidelines (EPRI 3002007572), EPRI Steam Generator In-Situ Pressure Test Guidelines (EPRI 1025132) and EPRI Steam Generator Integrity Assessment Guidelines (EPRI 3002007571).

   Degraded plugs, divider plate assemblies, tube-to-tubesheet welds, heads (interior surfaces), tubesheets (primary side), and secondary side internals are evaluated for continued acceptability on a case-by-case basis, as is leaving a loose part or a foreign object in a steam generator. NEI 97-06 and the associated EPRI guidelines provide guidance on the performance of some of these evaluations. The intent of all evaluations is to ensure that the components will continue to perform their functions, consistent with the design and licensing basis of the facility, and will not affect the integrity of other components (e.g., by generating loose parts).

Guidance on the acceptability of primary-to-secondary leakage and water chemistry parameters also are discussed in NEI 97-06 and the associated EPRI guidelines

7. ***Corrective Actions:*** Results that do not meet the acceptance criteria are addressed in the applicant's corrective action program under those specific portions of the quality assurance (QA) program that are used to meet Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B. The Appendix of the GALL Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the corrective actions element of this AMP for both safety-related and nonsafety-related structures and components (SCs) within the scope of this program.

   For degradation of steam generator tubes and sleeves (if applicable), the TSs provide requirements on the actions to be taken when the acceptance criteria are not met. For degradation of other components, the appropriate corrective action is evaluated per NEI 97-06 and the associated EPRI guidelines, the American Society of Mechanical Engineers (ASME) Code Section XI (2004 Edition),[13] 10 CFR 50.65, and 10 CFR Part 50, Appendix B, as appropriate.

8. ***Confirmation Process:*** The confirmation process is addressed through those specific portions of the QA program that are used to meet Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B. The Appendix of the GALL Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the confirmation process element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program.

   The adequacy of the preventive measures in the Steam Generator program is confirmed through periodic inspections.

9. ***Administrative Controls:*** Administrative controls are addressed through the QA program that is used to meet the requirements of 10 CFR Part 50, Appendix B, associated with managing the effects of aging. The Appendix of the GALL Report describes how an applicant may apply its 10 CFR Part 50, Appendix B, QA program to fulfill the administrative controls element of this AMP for both safety-related and nonsafety-related SCs within the scope of this program.

10. ***Operating Experience:*** Several generic communications have been issued by the NRC related to the steam generator programs implemented at plants. The reference section lists many of these generic communications. In addition, NEI 97-06 provides guidance to the industry for routinely sharing pertinent steam generator operating experience and for incorporating lessons learned from plant operation into guidelines referenced in NEI 97-06. The latter includes providing interim guidance to the industry, when needed.

    The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals) such that the steam generators can perform their intended safety function.

    The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience.

[13] Refer to the GALL Report, Chapter 1, for applicability of other editions of the ASME Code.

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**Revised GALL Report Tables IV.D1 and IV.D2**

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

| Item | Link | Structure and/or Component | Material | Environment | Aging Effect/ Mechanism | Aging Management Program (AMP) | Further Evaluation |
|---|---|---|---|---|---|---|---|
| IV.D1.RP-367 | IV.D1-6(RP-21) | Primary side components: divider plate | Steel (with nickel-alloy cladding); nickel alloy | Reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" A plant-specific program is to be evaluated for nickel alloy divider plate assemblies and associated welds made of Alloy 600; the effectiveness of the existing aging management programs should be verified to ensure that cracking due to PWSCC is not occurring if the conditions at the unit are not bounded by the industry analyses. | Yes, plant-specific. |
| IV.D1.RP-385 | | Tube-to-tube sheet welds | Nickel alloy | Reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" A plant-specific program is to be evaluated; the effectiveness of the existing aging management programs should be verified to ensure cracking is not occurring if the conditions at the unit are not bounded by the industry analyses and approval has not been granted permanently for relocating the reactor coolant pressure boundary function from the welds. | Yes, plant-specific. |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| IV.D1.R-436a | | Steam generator channel heads and tubesheets | Steel (with stainless steel or nickel alloy cladding) | Reactor coolant | Loss of material due to boric acid corrosion | AMP XI.M2, “Water Chemistry,” and AMP XI.M19, “Steam Generators” | No |

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

| Item | Link | Structure and/or Component | Material | Environment | Aging Effect/ Mechanism | Aging Management Program (AMP) | Further Evaluation |
|---|---|---|---|---|---|---|---|
| IV.D2.RP-185 | IV.D2-4(R-35) | Tube-to-tube sheet welds | Nickel alloy | Reactor coolant | Cracking due to primary water stress corrosion cracking | Chapter XI.M2, "Water Chemistry," and Chapter XI.M19, "Steam Generators" A plant-specific program is to be evaluated; the effectiveness of the existing aging management programs should be verified to ensure cracking is not occurring if the conditions at the unit are not bounded by the industry analyses and approval has not been granted permanently for relocating the reactor coolant pressure boundary function from the welds. | Yes, plant-specific |
| IV.D2.R-440a | | Steam generator upper and lower heads and tubesheets | Steel (with stainless steel or nickel alloy cladding) | Reactor coolant | Loss of material due to boric acid corrosion | AMP XI.M2, "Water Chemistry," and AMP XI.M19, "Steam Generators" | No |